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ENVIRONMENTAL ANALYSIS OF THE URANIUM FUEL CYCLE

PART II. NUCLEAR POWER REACTORS

Office of Radiation Programs

November 1973

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The principal purpose of this report is to project what effects the total uranium fuel cycle may have on public health and to indicate where, when, and how standards limiting environmental releases could be effectively applied to mitigate these effects.

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ENVIRONMENTAL ANALYSIS OF THE URANIUM FUEL CYCLE

PART II - Nuclear Power Reactors

November 1973

U. S. ENVIRONMENTAL PROTECTION AGENCY
Office of Radiation Programs
Technology Assessment Division
401 M Street, S.W.
Washington, D.C. 20460

FOREWORD

The generation of electricity by light-water-cooled nuclear power reactors using enriched uranium for fuel is experiencing rapid growth in the United States. This increase in nuclear power reactors will require similar growth in the other activities that must exist to support these reactors. These activities, the sum total of which comprises the uranium fuel cycle, can be conveniently separated into three parts: 1) the operations of milling, conversion, enrichment, fuel fabrication and transportation that convert mined uranium ore into reactor fuel, 2) the light-water-cooled reactor that burns this fuel, and 3) the reprocessing of spent fuel after it leaves the reactor.

This report is one part of a three-part analysis of the impact of the various operations within the uranium fuel cycle. The complete analysis comprises three reports: The Fuel Supply (Part I), Light-Water Reactors (Part II), and Fuel Reprocessing (Part III). High-level waste disposal operations have not been included in this analysis since these have no planned discharges to the environment. Similarly, accidents, although of potential environmental risk significance, have also not been included. Other fuel cycles such as plutonium recycle, plutonium, and thorium have been excluded. Insofar as uranium may be used in high-temperature gas-cooled reactors, this use has also been excluded.

The principal purposes of the analysis are to project what effects the total uranium fuel cycle may have on public health and to indicate where, when, and how standards limiting environmental releases could be effectively applied to mitigate these effects. The growth of nuclear energy has been managed so that environmental contamination is minimal at the present time; however, the projected growth of this industry and its anticipated releases of radioactivity to the environment warrant a careful examination of potential health effects. Considerable emphasis has been placed on the long-term health consequences of radioactivity releases from the various operations, especially in terms of expected persistence in the environment and for any regional, national or world-wide migration that may occur. It is believed that these perspectives are important in judging the potential impact of radiation-related activities and should be used in public policy decisions for their control.

Comments on this analysis would be appreciated. These should be sent to the Director, Criteria and Standards Division of the Office of Radiation Programs.

W D Rowe

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Deputy Assistant Administrator
for Radiation Programs

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PART II. NUCLEAR POWER REACTORS

INTRODUCTION

Present estimates of electrical power growth indicate a substantial increase in the growth of nuclear powered generating stations. By the year 2000 approximately 65 per cent of the U.S. electrical generation is expected to come from nuclear energy. In order to meet this projected demand, approximately 1200 nuclear reactors with a capacity of one-gigawatt each (1 GWe = 1,000,000 kilowatts) will be required. Projections of future technology indicate that the Liquid-Metal Fast Breeder Reactor (LMFBR) is expected to account for a substantial portion of this forecasted capacity. Based upon these projections only about 500 GWe will be from light-water-cooled reactors. However, for the purposes of this analysis, all nuclear power stations installed through the year 2000 are assumed to employ light-water reactors.

The capacity of individual reactors has increased from 50-200 MWe in the early 1960's to 1100-1200 MWe (1.1-1.2 GWe) for advanced reactors presently being ordered by utilities. Problems associated with emergency core cooling methods have led to a reduction in the permissible operating power density and, consequently, 1000 MWe has been assumed as a reference power level for this analysis. If these core-cooling problems are resolved, it is possible that the trend toward larger units will continue so that reactors installed in the latter part of this century might be considerably larger than the 1000 MWe size assumed in this study.

There are two basic types of light-water reactors: the pressurized-water direct cycle plant (PWR) and the indirect cycle boiling water reactor (BWR). The method of operation and the differences between these two types will be discussed in the following section. At present there are three domestic manufacturers of the pressurized-water type: the Westinghouse Electric Corporation, the Babcock & Wilcox Company, and the Combustion Engineering Corporation. There is only one domestic manufacturer of the boiling-water reactor, the General Electric Company. At the present time, pressurized-water reactors comprise approximately two-thirds of the light-water generating capacity committed through 1982. This 2:1 PWR:BWR ratio has been assumed to continue through the year 2000.

PROCESS DESCRIPTION

A light-water-cooled nuclear power station operates on the same principle as a conventional fossil-fueled (oil or coal) power station except that the heat generation is by nuclear fission rather than combustion. The heat liberated in either process is used to convert water into steam. The steam enters a three-stage turbine consisting of one high - pressure stage and two low-pressure stages. The turbine consists of a common central shaft attached to a circular array of curved blades. The steam impacts on these blades turning the rotor at high speeds. The turbine shaft is connected to a wire wound armature in the generator. This armature rotates in an applied magnetic field producing alternating electric current.

After passing through the turbine the low pressure steam passes through a condenser where the steam transfers its remaining heat to the condenser cooling water and is condensed back into water and is recycled into the boiler. The heated condenser cooling water may be released directly to the environment in a single-pass open-cycle cooling system. However, this heated water may have an adverse impact on aquatic organisms and the use of open-cycle systems is decreasing in favor of augmented cooling systems. This is particularly important for nuclear power plants as they have a lower thermal efficiency (32%) than fossil-fueled plants (40%) and, consequently, discharge about two-thirds of their heat output to the environment.

There are several types of auxiliary cooling systems which have been proposed for nuclear power plants. The open cycle system may be retained with the addition of evaporation ponds, long discharge canals, or spray canals which permit the excess heat to be transmitted to the atmosphere prior to discharge of the condenser cooling water to the receiving water body. The other principal alternative is to employ a closed-cycle cooling system which transfers the heat almost completely to the atmosphere using cooling towers and recycles the cooled water back to the condenser.

The principle differences between the nuclear and conventional electric generating stations are in the type and quantity of fuel consumed and the nature of the residuals which are discharged from the process. The fossil-fueled plant will produce sulfur oxides, carbon monoxide, nitrogen oxides, hydrocarbons, and particulate (dust) emissions. The nuclear power plant produces highly radioactive atoms from the fission of the uranium atoms (fission products) and also from the absorption of neutrons by the coolant and structural materials (activation products). These radioactive materials are largely contained within the reactor fuel elements. The greatest danger from a nuclear power plant would be the release of significant quantities of these materials as a consequence of fuel element melting in a serious accident. Because of the enormous quantity of radioactive material generated in a nuclear power plant,and the inherent hazard associated with this material, many precautions must be taken to insure that these materials are not released to the environment. One of the most visible precautions against the accidental release of radioactive materials is the reactor containment which provides for total enclosure of the reactor and most of the principal reactor systems. This containment structure is usually cylindrical but may be enclosed within another building. A sketch of a typical two-unit nuclear power station is shown in Figure 1. This illustration depicts the principal structures and features which are externally visible and characteristic of a nuclear power plant.

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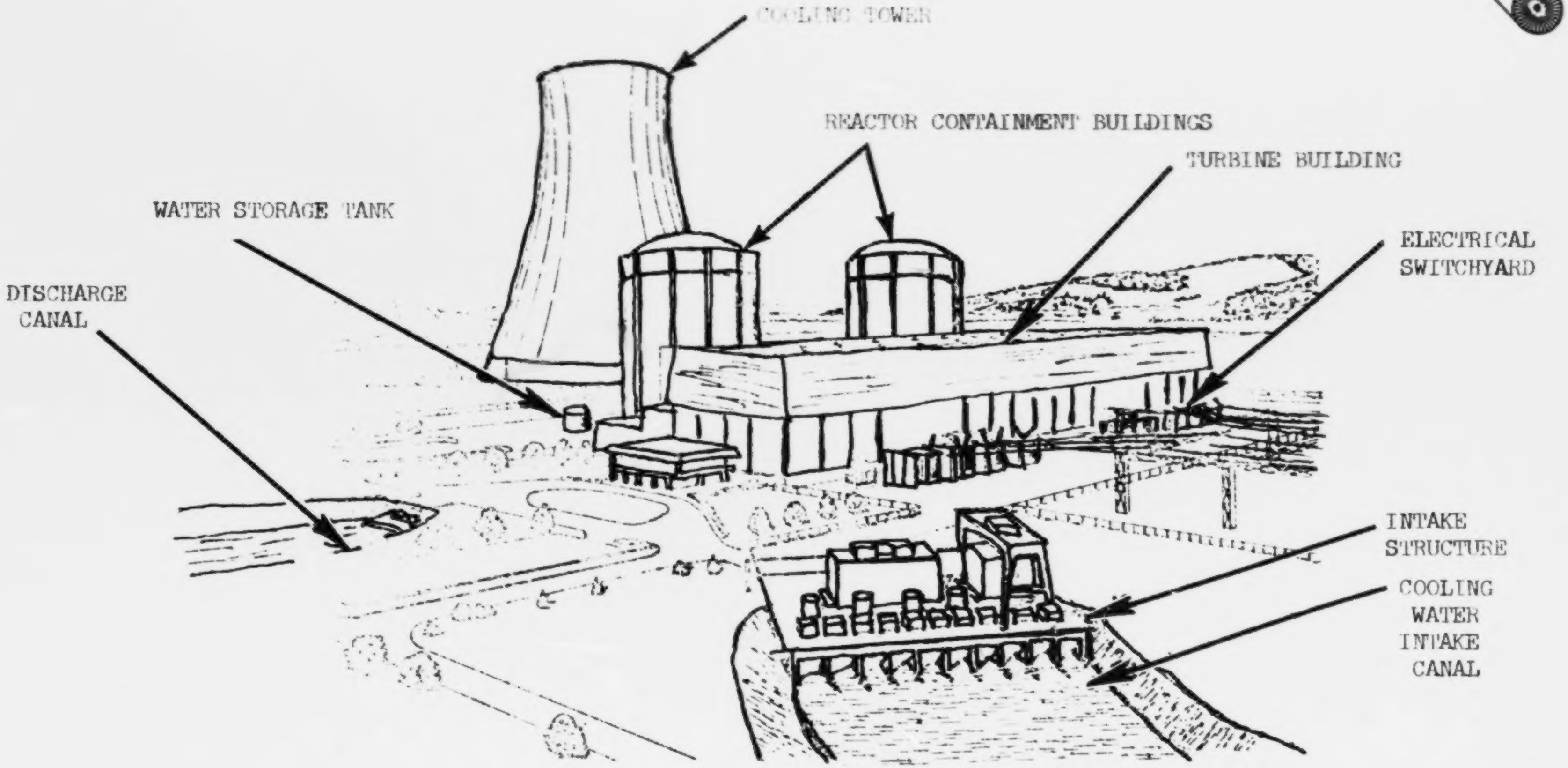

COOLING TOWER
REACTOR CONTAINMENT BUILDINGS
TURBINE BUILDING
WATER STORAGE TANK
DISCHARGE CANAL
ELECTRICAL SWITCHYARD
INTAKE STRUCTURE
COOLING WATER INTAKE CANAL

Figure 1. Appearance of a Typical Light-Water-Cooled Nuclear Power Station Site

Light-Water Nuclear Reactor Types

There are two basic types of light-water-cooled nuclear reactors: the pressurized-water reactor or indirect cycle and the boiling water reactor which operates on a direct cycle. The fundamental difference between these two designs is evident from a comparison of the two reactor systems which are shown in Figures 2 and 3. In the pressurized-water reactor (Figure 2), the coolant is maintained at a high pressure (∿2250 pounds per square inch) which inhibits boiling. Steam is produced by allowing the heated primary coolant to transfer heat to a secondary coolant which is at a lower pressure (∿1000 psi) where boiling can occur. Because the steam production is separated from the heat generation source, this mode of operation is termed an indirect cycle.

The boiling water reactor operates on a direct cycle where the process steam is generated directly in the reactor vessel. This is possible because of a lower reactor coolant pressure (∿1020 psi) than in the pressurized-water reactor. The steam generated in the reactor vessel is separated from excess moisture and passes directly to the first stage (high-pressure) turbine. The principal components of the boiling-water reactor system are shown in Figure 3.

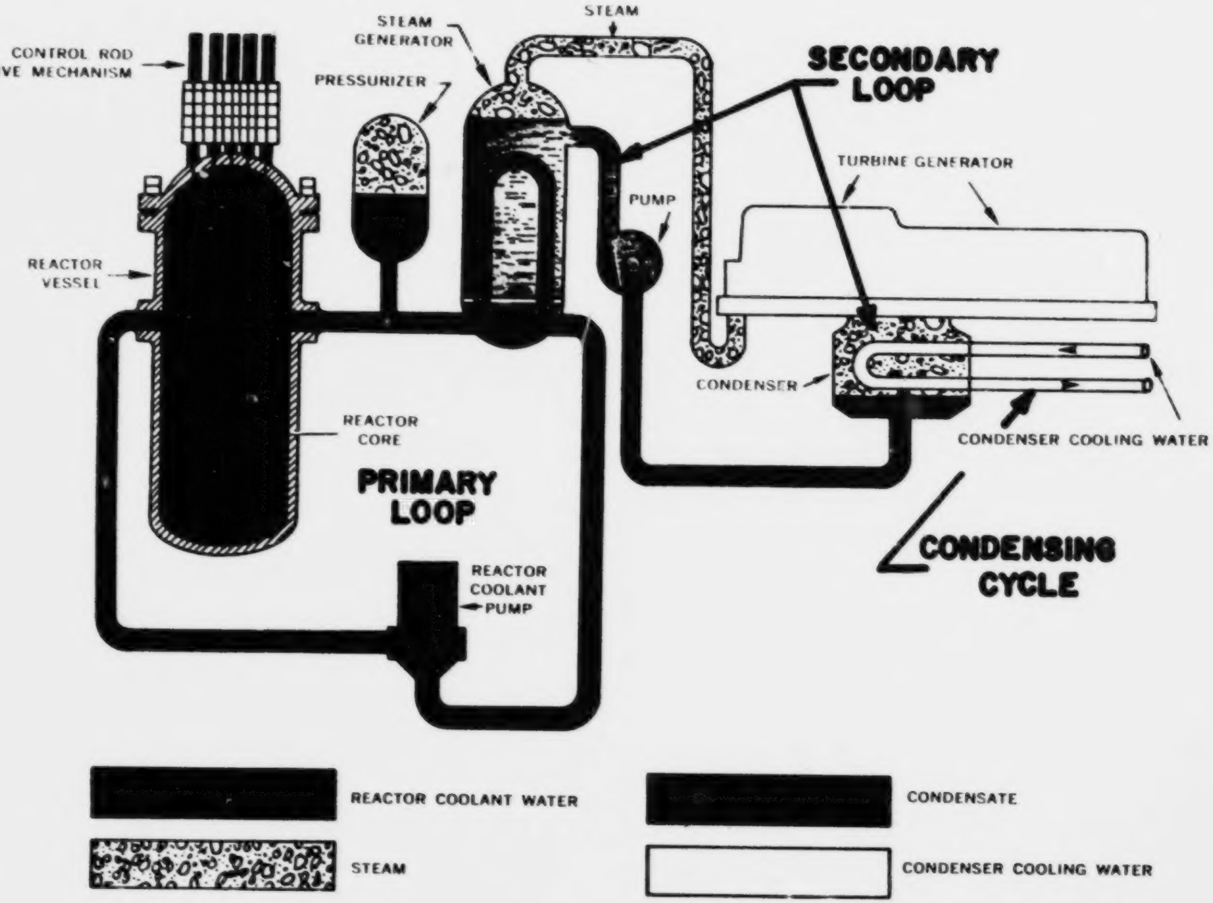

CONTROL ROD DRIVE MECHANISM
REACTOR VESSEL
PRESSURIZER
STEAM GENERATOR
STEAM
SECONDARY LOOP
TURBINE GENERATOR
PUMP
CONDENSER
CONDENSER COOLING WATER
REACTOR CORE
PRIMARY LOOP
REACTOR COOLANT PUMP
CONDENSING CYCLE
REACTOR COOLANT WATER
CONDENSATE
STEAM
CONDENSER COOLING WATER

Figure 2. Pressurized-Water Reactor Schematic

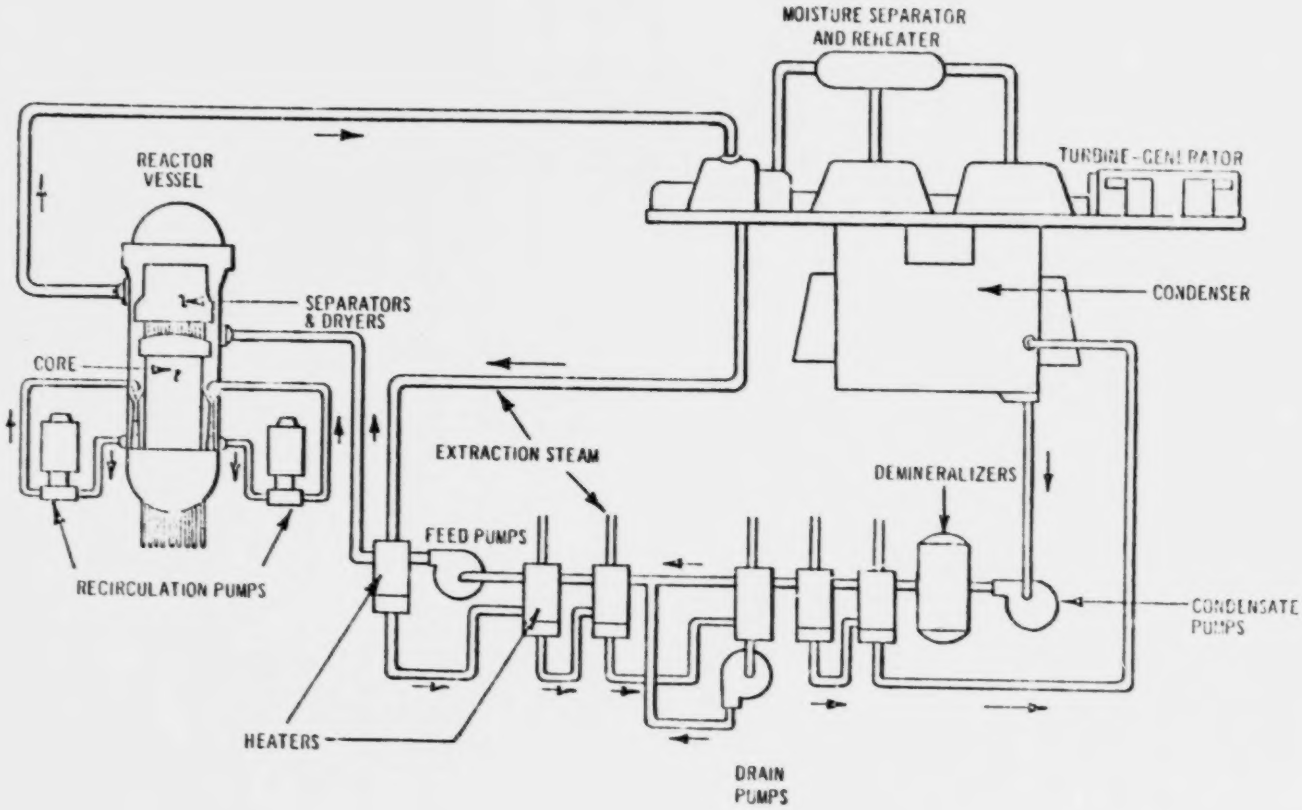

MOISTURE SEPARATOR AND REHEATER
REACTOR VESSEL
TURBINE-GENERATOR
SEPARATORS & DRYERS
CONDENSER
CORE
EXTRACTION STEAM
DEMINERALIZERS
FEED PUMPS
RECIRCULATION PUMPS
CONDENSATE PUMPS
HEATERS
DRAIN PUMPS

Figure 3. Boiling Water Reactor Schematic

Site Characteristics

The local radiological impact of any nuclear power plant effluent is strongly affected by site related characteristics which govern dispersion, reconcentration, and other environmental transport mechanisms. The integrated population dose is also governed by demographic characteristics such as location of population centers and average population density. In order to incorporate these considerations into the assessment of the radiological impact, three representative sites were considered. The selection of these sites was based upon the nature of the body of water receiving waste discharges, population density, and meteorological conditions. The three sites include one site from each of three major classifications of site locations: seacoast, river and lake. The demographic characteristics of these sites are presented in Table 1, which is based on the enclosed population values of 50 existing sites. Annual average atmospheric dispersion factors for locations of particular interest are also given in Table 1.

The liquid effluent dilution factors are calculated on a conservative basis for both the maximum individual and average individual in the population. Values for the river site dilution for the maximum individual and the general population is based upon the ratio of the average condenser cooling water flow rate to the river flow rate. For a 1,000 MWe BWR or PWR the assumed condenser flow is approximately 1,800 cubic feet per second. The assumed river flow rate was 50,000 cubic feet per second. This dilution would be less for augmented cooling systems such as cooling towers, but at large distances from the discharge canal, this difference does not remain appreciable.

Table 1 also presents estimates of the atmospheric dispersion factors which relate airborne concentration to the rate of release from the reactor. These are calculated for two release heights, ground level and a 100-meter stack. In order to assure that these dispersion factors for the selected sites were truly representative, additional sites were examined. These additional sites are listed in Table 2. The number of plants examined in each site category was chosen on the basis of the estimated mix of site locations. The values for the three representative sites appear to be typical for the average sites except for the river site where the dispersion conditions are more favorable at the selected site than at the majority of river sites. The consequences of this difference will be explored in the dose evaluation section.

Table 1

Principal Characteristics of Representative Reactor Sites

Site Location	Seacoast	River	Lake
		Enclosed Population	
< 5 miles	4,300	30,883	1,439
< 10 miles	5,914	339,704	25,787
< 20 miles	231,729	883,774	103,206
< 50 miles	3,517,236	6,528,988	749,884
Distance (miles) to:			
Site Boundary	0.5	0.5	0.5
Nearest Resident	.75	.73	.5
Nearest Farm	4.5	3.5	2.5
Annual Average Atmospheric Dispersion Factors (χ/Q, seconds/cubic meter)			
10-meter vent release			
Site Boundary	1.2×10^{-6}	2.9×10^{-6}	1.7×10^{-6}
Nearest Resident	6.0×10^{-7}	1.7×10^{-6}	1.7×10^{-6}
Nearest Farm	3.3×10^{-8}	1.5×10^{-7}	1.4×10^{-7}
100-meter stack release			
Site Boundary	3.6×10^{-8}	3.3×10^{-7}	2.7×10^{-8}
Nearest Resident	3.2×10^{-8}	2.2×10^{-7}	2.7×10^{-8}
Nearest Farm	1.8×10^{-8}	2.3×10^{-8}	2.8×10^{-8}

Table 2

Atmospheric Dilution Factors For Vent Release (h=0) at Site Boundaries

Site	Plant	Annual Average χ/Q (Ci/m^3)/(Ci/sec)	Distance & Direction
Seacoast	Turkey Point	3.0×10^{-6}	640 m NE
	Forked River	1.0×10^{-5}	640 m SE
	Calvert Cliffs	2.7×10^{-6}	1190 m SE
Seacoast Average (14%)		5.2×10^{-6}	
River	Indian Point 2	2.6×10^{-6}	1000 m S
	Arkansas Nuclear One	4.4×10^{-6}	1040 m W
	Maine Yankee	1.2×10^{-5}	610 m NNE
	Monticello	7.8×10^{-6}	750 m SSE
	Salem	5.0×10^{-5}	190 m WSW
	Cooper	8.0×10^{-5}	122 m NE
	Hanford 2	1.2×10^{-6}	1950 m SE
	Three Mile Island	9.1×10^{-6}	660 m ESE
	Fort Calhoun	2.2×10^{-5}	400 m ESE
	V. C. Summer	3.6×10^{-6}	1600 m E
	W. H. Zimmer	9.6×10^{-6}	~170 m W
	LaSalle	8.0×10^{-6}	190 m W
	North Anna	$\sim 2.8 \times 10^{-6}$	1500 m SE
	Waterford	2.2×10^{-5}	320 m NNE
River Average (67%)		1.7×10^{-5}	
Lake	Fitzpatrick	4.0×10^{-7}	960 m SW
	Zion	3.8×10^{-6}	320 m N
	Kewaunee	2.4×10^{-6}	~450 m ESE
	Fermi 2	1.2×10^{-6}	~1000 m S
Lake Average (19%)		2.0×10^{-6}	
Grand Average		1.2×10^{-5}	750 m

Source:
From U. S. Atomic Energy Commission Final Environmental Statements for these respective plants.

SOURCES OF RADIOACTIVE DISCHARGES TO THE ENVIRONMENT

Nuclear power reactors generate radioactive materials as a consequence of the fissioning of uranium and by neutron absorption in the coolant and in structural materials which leads to induced radioactivity in these components. The products of uranium fission comprise a large number of elements and include both stable and radioactive isotopes of these elements. Among the more important radionuclides produced by uranium fissioning are isotopes of the noble gases krypton and xenon, the alkali metals cesium and rubidium, the alkaline earths barium and strontium, and the halogens iodine and bromine.

The capture of the neutrons liberated in fission by the nuclei of stable elements often results in the production of radioactive activation products. The coolant activation products are generally gases such as argon-41, fluorine-18, nitrogen-13, nitrogen-16, and oxygen-19 which have short half-lives in the range of several seconds to a few hours. The induced activities in the structural materials may have considerably longer half-lives and comprise a much wider range of elements including zirconium, manganese, nickel, iron, carbon, chromium, cobalt, and copper. These radionuclides usually remain fixed in the structural materials but can enter the coolant as a consequence of corrosion and erosion in the pumps and other moving components.

Nuclear power reactors are constructed with multiple barriers for isolating these radionuclides from the environment. The principal barriers are: (1) the fuel cladding, (2) the reactor systems, and (3) the reactor and auxiliary buildings. Release of radioactive material to the environment occurs principally as a consequence of the penetration of one or more of these barriers. This penetration can occur due to the presence of structural defects, leakage from pumps or other components, or intentionally as a consequence of the particular plant design. The remainder of this section discusses the specific release pathways from each barrier while providing typical estimates for the magnitude of release from each pathway.

Fuel Cladding Defects

The primary barrier for isolating radioactive fission products from the environment is the fuel rod cladding. Within a 1000-MWe nuclear reactor there are millions of curies of radioactive isotopes; the iodine-131 alone can amount to over 70 megacuries. The major fraction of these fission products is retained within the ceramic matrix of the uranium dioxide fuel pellet. However, the more volatile elements such as the halogens and noble gases can diffuse through this matrix into the space between the fuel pellet and the cladding. This diffusion process is accelerated by the high fuel temperatures and the presence of cracks and fissures produced by thermal stresses so that appreciable amounts of these elements accumulate in the fuel-cladding gap. Nonvolatile elements such as strontium, barium, and cerium also accumulate there as they are daughter products produced by the radioactive decay of short-lived noble gas precursor radionuclides. These radionuclides will be contained within the fuel rod as long as the thin metallic cladding remains intact. In this situation, the quantity of radioactive material reaching the coolant will be limited to fission products arising from small traces of uranium which remain on the outer surface of the fuel as a consequence of the manufacturing process, activation products which arise from neutron induced reactions with water and air, and traces of metallic elements which enter the coolant as a result of corrosion of the reactor vessel, piping,and other structural components.

The fuel cladding is approximately twenty-five thousandths of an inch in thickness for PWRs and approximately thirty-three thousandths of an inch for BWRs and is subjected to thermal stresses as the reactor power level is changed and mechanical stresses from the high pressure and velocity of the coolant or from physical contact with the fuel as it expands. These stresses, combined with variations in the cladding thickness or other irregularities in manufacture, can result in small pin-holes or defects in the fuel cladding which allow the volatile radionuclides in the cladding gap to escape into the coolant. Under severe conditions large failures could occur in the cladding which would permit the coolant to contact the fuel and leach out the less volatile fission products. However, these occurrences are not common and cladding failures of the small pin-hole type are more usual.

The extreme conditions imposed in the reactor core on the fuel cladding together with the difficulties of producing large quantities of thin, near-perfect tubing for the large number of fuel rods (approximately 40,000) make it extremely difficult to eliminate such fuel cladding failures. As a result, nuclear reactor systems are designed to accommodate the equivalent of 1 percent of the gap activity (contained in all of the fuel rods) escaping to the coolant through cladding defects. Table 3 shows the relationships between the total core inventory, the fuel plenum (gap) inventory, and the primary coolant inventory for a representative 1000-MWe light-water reactor. Even with defective fuel, the primary coolant activity

Table 3

Volatile Radionuclide Inventory in a 1000 MWe Nuclear Power Plant

Parameters: 3040 MWt PWR, operating at full power for 500 days with 1% of the fuel rods having cladding defects.

Radionuclide	Half-life	Total Activity In: Reactor Core (megacuries)[a]	Fuel-Cladding Gap (megacuries)[a]	Primary Coolant (curies)
Iodines				
I-131	8.05d	74.9	0.76	465
I-132	2.3 h	114.0	0.14	186
I-133	21. h	171.0	0.64	766
I-134	52. m	206.0	0.12	117
I-135	6.7 h	158.0	0.34	420
Kryptons				
Kr-85	10.8 y	0.66	0.067	334
Kr-85m	4.4 h	33.5	0.95	439
Kr-87	76. m	64.4	0.076	261
Kr-88	2.8 h	93.0	0.149	775
Xenons				
Xe-133	5.3 d	164.0	4.17	52,290
Xe-133m	2.3 d	4.0	0.019	692
Xe-135	9.2 h	43.6	0.084	1,488
Xe-135m	15.6 m	46.4	0.016	42

[a] 1 megacurie = 1,000,000 curies (10^6 curies)

Source and Assumptions: Appendix D of the Final Safety Analysis for the Kewaunee Nuclear Power Plant (converted from 1721.4 MWt to 3040 MWt and core volume adjusted to scale).

remains a small fraction of the total inventory within the reactor. The coolant purification systems are responsible for removing most of the primary coolant activity so that low levels are maintained in the circulating coolant. Typical primary coolant radionuclide concentrations for a reactor having 1 percent failed fuel are shown in Table 4.

Table 4

Estimated Reactor Coolant Specific Fission Product and Corrosion Product Activities (at 578° F)

Fission product reactor coolant concentrations corresponding to 1% Failed Fuel

Isotope	µCi/cc
Noble Gas Fission Products	
Kr-85	1.11
Kr-85m	1.46
Kr-87	0.87
Kr-88	2.58
Xe-133	1.74×10^{2}
Xe-133m	1.97
Xe-135m	0.14
Xe-138	0.36
Total Noble Gases	187.3
Corrosion Products	
Mn-54	4.2×10^{3}
Mn-56	2.2×10^{-2}
Co-58	8.1×10^{-3}
Fe-59	1.8×10^{-3}
Co-60	1.4×10^{-3}
Total Corrosion Products	3.7×10^{-2}

Isotope	µCi/cc
Fission Products	
Br-84	3.0×10^{-2}
Rb-88	2.56
Rb-89	6.7×10^{-2}
Sr-89	2.52×10^{-3}
Sr-90	4.42×10^{-5}
Y-90	5.37×10^{-5}
Y-91	4.77×10^{-4}
Sr-92	5.63×10^{-4}
Y-92	5.54×10^{-4}
Zr-95	5.04×10^{-4}
Nb-95	4.70×10^{-4}
Mo-99	2.11
I-131	1.55
Te-132	0.17
I-132	0.62
I-133	2.55
Te-134	2.2×10^{-2}
I-134	0.39
Cs-134	7.0×10^{-2}
I-135	1.4
Cs-136	0.33
Cs-137	0.43
Cs-138	0.48
Ce-144	2.3×10^{-4}
Pr-144	2.3×10^{-4}
Total Fission Products	12.8

Source: Kewaunee Final Safety Analysis Report, Appendix D, Table D 4-1

BWR Condenser Air Ejector Off-gas

The boiling-water reactor operates on a direct cycle and the contaminated coolant passes directly through the turbine. Entrained radioactive gases, air which has leaked into the condenser, and hydrogen and oxygen which result from the radiolytic dissociation of water are removed from the main turbine condenser by the steam jet air ejector which is used to maintain a vacuum in the condenser. These gases are removed at a rate of about 300 cubic feet per minute. Approximately 230 cfm represents the dissociated hydrogen and oxygen, 5 - 20 cfm represents air in-leakage and the remainder in water vapor; the radioactive gases contributing negligible volume.

In the absence of appreciable failed fuel, the principal contributor to the radioactive emission is nitrogen-13. When there is significant failed fuel, the noble gas fission product releases dwarf the activation gas releases as shown in Table 5.

PWR Gaseous Radwaste System

In the operation of a PWR, boron is added to the primary coolant to act as a neutron absorber. In the beginning of the fuel cycle its concentration is approximately 1,000 ppm. As the reactor produces power, less and less boron is required. In order to remove this boron a small portion of the coolant purification flow is typically "bled" to the boron recovery system (Figure 5). Radioactive gases evolved at the gas stripper are routed to the waste gas system for treatment. Table 6 provides an estimate of the radioactivity releases from a waste gas system providing 45 days of holdup for these gases.

Table 5

Estimated Air Ejector Off-gas Release Rates Following 30-Minute Holdup 1064-MWe BWR with 0.25% Failed Fuel, 18.5 scfm in-leakage

Radionuclide	Half-life	Emission rate µCi/sec	Annual Discharge Ci/yr
Nitrogen-13	10 min	340	8,580
Krypton-83m	1.9 hrs	2,537	64,000
Krypton-85m	4.4 hrs	5,700	143,800
Krypton-85	10.8 yrs	7.5	189
Krypton-87	76 min	15,700	396,000
Krypton-88	2.8 hrs	17,367	438,000
Krypton-89	3.2 min	262	6,610
Xenon-131m	11.8 days	15	378
Xenon-133m	2.3 days	188	4,743
Xenon-133	5.3 days	5,100	128,700
Xenon-135m	9.1 hrs	6,000	202,000
Xenon-135	15.6 min	17,367	438,150
Xenon-137	3.9 min	860	21,700
Xenon-138	17.5 min	26,500	668,600
Total		100,000	2,523,000

Source: Browns Ferry Final Safety Analysis Report.

Table 6

Representative Estimated Gaseous Releases Associated with Primary-to-Secondary Leakage (20 Gallons per Day)

Principal Radionuclide	Annual Activity Release to the Environment (curies per year) from Containment Purge	Waste Gas Processing System	Steam Generator Leakage
Krypton-85	13.0	791	2.0
Krypton-87	0.04	-	3.0
Krypton-88	-	-	10.0
Xenon-131m	10.0	63	3.0
Xenon-133	1005.0	1500	682.0
Xenon-135	0.018	-	3.0
Xenon-138	0.007	-	2.0
Iodine-131	0.018	-	0.62

Source: (Table III-3) of the AEC Draft Environmental Impact Statement for Indian Pt 2

Liquid Radioactive Waste Treatment Systems

In BWRs and PWRs, various sources of liquid waste are handled by liquid waste treatment systems. Each reactor type provides for the purification of the reactor coolant. In BWRs this system is simply referred to as the reactor water cleanup system (RWCS) and is shown in Figure 4. In PWRs, coolant purification (and chemical adjustment) is provided by the chemical and volume control system (CVCS) which itself may be classified into two subsystems, the reactor coolant cleanup subsystem and the boron recovery subsystem (Figure 5). The boron recovery subsystem of the CVCS in a PWR, and RWCS in a BWR, may contribute radioactive liquids to the respective liquid radioactive waste treatment systems in each type of reactor. These liquid radioactive waste treatment systems handle the miscellaneous radioactive liquids generated by plant operation as well as those liquids from the coolant purification systems. Figures 7 and 11 (pages 43 and 53) illustrate liquid radwaste systems representative of BWRs and PWRs presently operating. Table 7 (page 28) compares the magnitude of PWR liquid radioactivity releases from the CVCS, the liquid radwaste system, and steam generator blowdown during a postulated 20 gallon per day primary-to-secondary leak.

Primary-to-Secondary Leakage in PWRs

In pressurized water reactors, the secondary coolant system is isolated from the primary reactor coolant by virture of the tubing in the steam generators. If these tubes remain intact, the secondary system would be free of radioactive material. The number of these tubes can be about 4000 per steam generator, depending upon the reactor make and plant design power level, and the number of

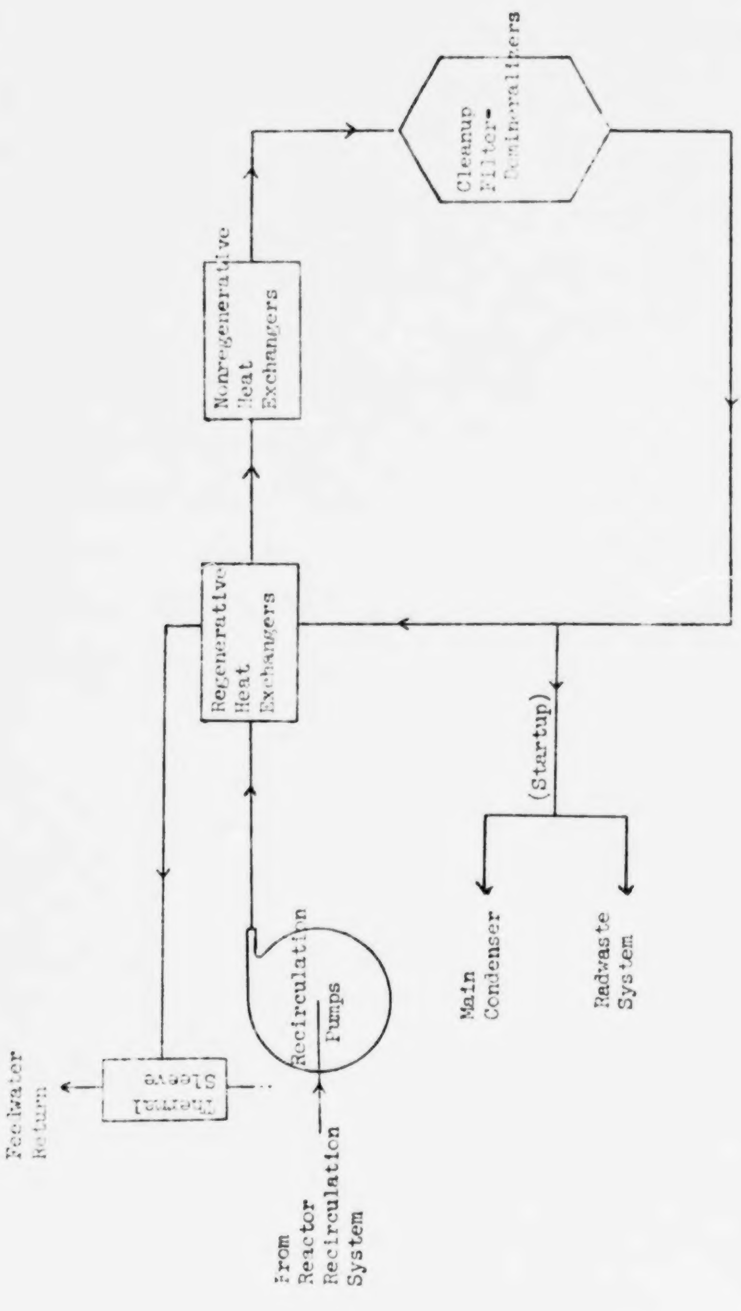
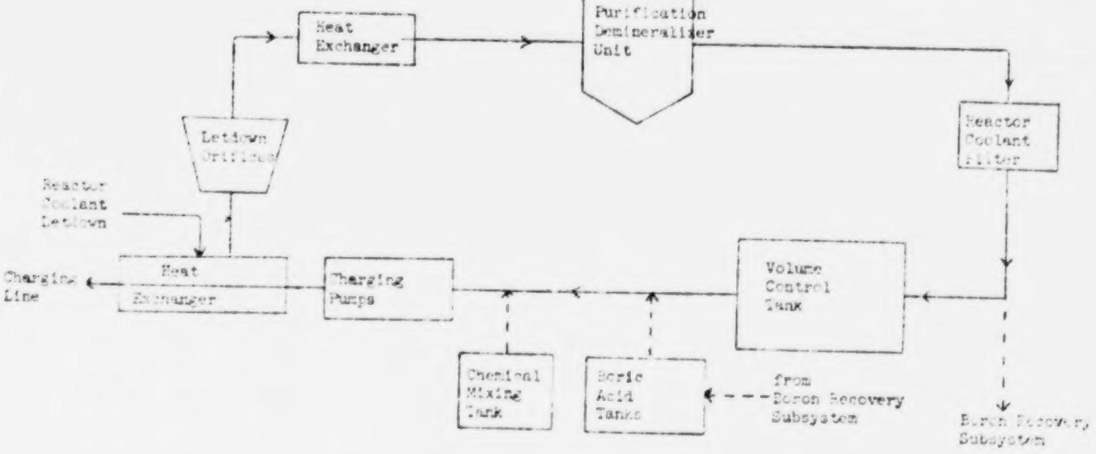

Cleanup Filter-Demineralizers
Nonregenerative Heat Exchangers
Regenerative Heat Exchangers
(Startup)
Main Condenser
Radwaste System
Recirculation Pumps
Thermal Sleeve
Feedwater Return
From Reactor Recirculation System

Figure 4. BWR Reactor Water Cleanup System

Reproduced from best available copy.

Reactor Coolant Cleanup Subsystem


Heat Exchanger
Purification Demineralizer Unit
Letdown Orifices
Reactor Coolant Filter
Reactor Coolant Letdown
Charging Line
Heat Exchanger
Charging Pumps
Volume Control Tank
Chemical Mixing Tank
Boric Acid Tanks
from Boron Recovery Subsystem
Boron Recovery Subsystem

Boron Recovery Subsystem

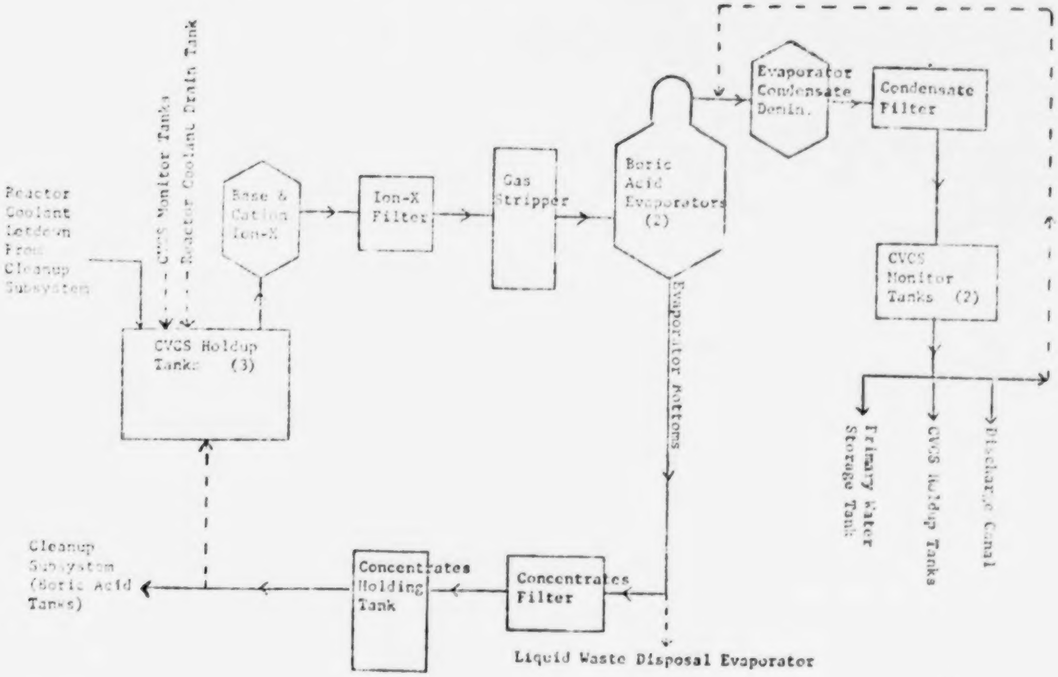

Reactor Coolant Letdown From Cleanup Subsystem
CVCS Monitor Tanks
Reactor Coolant Drain Tank
Base & Cation Ion-X
Ion-X Filter
Gas Stripper
Boric Acid Evaporators (2)
Evaporator Condensate Demin.
Condensate Filter
CVCS Monitor Tanks (2)
CVCS Holdup Tanks (3)
Evaporator Bottoms
Primary Water Storage Tank
CVCS Holdup Tanks
Discharge Canal
Cleanup Subsystem (Boric Acid Tanks)
Concentrates Holding Tank
Concentrates Filter
Liquid Waste Disposal Evaporator

Figure 5. PWR Chemical and Volume Control System (CVCS)

steam generators per plant may range between 2 and 4, depending on the power rating and reactor vendor. Due to the large number of tubes present in a plant, the possibility of defects in the tubing either due to manufacturing errors, or operating conditions (corrosion, burn-out, or stress) is enhanced. If holes develop in the steam generator tubing, the primary reactor coolant water will leak into the steam generator as a consequence of the higher primary system pressure. This water will contain radioactive materials at the concentration present in the primary coolant and consequently contaminate the secondary coolant system.

Volatile radionuclides which enter the secondary coolant system as a result of primary-to-secondary leakage may be discharged to the environment via two pathways: (1) the condenser air ejector and (2) the steam generator blowdown flash tank. The condenser air ejector removes entrained gases from the secondary coolant and will extract radioactive noble gases, gaseous activation products, and some of the halogens (iodines). In many designs, the steam is generated from evaporation of water in the steam generator. Solids build up in the steam generator and may impair heat transfer. To counteract this buildup, typically 5-15 gallons of the steam generator bottom liquid are withdrawn per minute. This hot liquid is bled into a flash tank at a lower pressure than the secondary coolant system. The low pressure and high temperature cause the rapid evaporation (flashing) of 30-40 percent of the liquid into steam which, in older plant designs, was released to the atmosphere. The volatile radionuclides would also be carried over in the steam and, consequently, released.

The relative magnitude of these gaseous releases from the steam generator leakage are shown in Table 6. The major contribution from these releases is the additional iodine-131 which can be considerably greater than that from other sources.

In older plants, the blowdown liquid remaining in the flash tank would be discharged to the condenser coolant without any cleanup. Any radionuclides which were in the steam generator blowdown as a result of primary-to-secondary leakage would be released to the environment. If appreciable (20 gallons per day) primary-to-secondary leakage is present, concurrent with discernible fuel cladding perforations, the unprocessed steam generator blowdown could be the major source of liquid radionuclide discharges as illustrated in Table 7.

Operating experience has shown that pressurized water reactors eventually develop some primary-to-secondary leakage. Generally, with only a few tubes having defects, this leakage may amount to only a few gallons per day. However, several plants have experienced long periods of operation with leakage rates of 50 gallons per day or more (Table 8). Because of the additional solids contributed from the boric acid in the primary coolant, high leakage rates cannot be tolerated for more than a few days. These high leakage rates require corrective action which usually means plugging up the defective tubes by sealing them off with plugs and small explosive charges or by welding.

Table 7

Comparison of Estimated Radionuclide Liquid Discharges to the Environment with Appreciable Primary-to-Secondary Leakage (20 Gallons per Day Leakage and 10 Gallons per Minute Blowdown)

Major Radionuclides	Activity (curies per year) Discharged from Chemical and Volume Control System	Waste Treatment System	Steam Generator Blowdown
Molybdenum-99	0.005	0.018	5.51
Technetium-99m	0.004	0.016	0.61
Iodine-130	0.002	0.006	0.009
Iodine-131	0.59	2.06	8.1
Iodine-132	0.056	0.19	0.12
Iodine-133	0.56	1.92	3.46
Cesium-134	0.004	-	7.1
Iodine-135	0.14	0.45	0.62
Cesium-136	0.001	0.005	2.05
Cesium-137	0.003	0.012	6.06
Totals	∿1.4	∿4.7	∿35

Source: Table III-4 of the AEC Draft Environmental Statement for the Indian Point Nuclear Station Unit 2.

Table 8

Primary-to-Secondary System Leakage Experience in Pressurized Water Reactors

Plant	Average Leakage Rate (gallons) per day)	Duration
H. B. Robinson Unit 2	55 14,400	7 months <1 day
Point Beach Unit 1	up to 50	several months
Connecticut Yankee (Haddam Neck)	1,500	several days
San Onofre Unit 1	up to 15 up to 95	several months several weeks
Yankee Rowe	up to 1,200	several months

Source: Operating reports for these respective plants.

System Leakage to Building Atmosphere

Release of radioactivity from reactor and waste treatment systems can occur via leakage directly from system components. Much of this leakage is from coolant pump or valve seals and is generally returned directly to the reactor coolant system. Other leakage paths include smaller valve seals and releases associated with chemical and radiological analysis. Most of the liquid released will be collected by plant drain systems and be processed by the waste treatment system. The volatile elements, including the noble gases and halogens, will be released to plant buildings such as the containment building where they are available for leakage or discharge to the environment. In pressurized water reactors, this leakage may average between 0.2 - 0.3[1] gallons per minute and account for 0.4 - 1.0% of the coolant volume per day. Similar leakage rates may be expected at boiling water reactor plants. Thus appreciable quantities of the volatile elements may accumulate in the reactor containment building and auxiliary building atmospheres.

Containment Purging

Radioactive halogens and noble gases which escape to the reactor containment from the reactor system may be discharged to the environment during containment venting or purging. The containment atmosphere is vented or purged in order to test the containment isolation system on a periodic basis, reduce containment temperature and activity levels prior to and during maintenance involving entry into the containment, and also to reduce containment pressure if excessive system leakage exists. The purges for testing are typically of one to several minutes in duration and may occur on a monthly schedule. The other purging intervals may last several hours or more and may occur 1-10 times per year. Estimated releases of gaseous radionuclides via containment purging are shown in Tables 8 and 9 for the two types of light-water reactors.

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[1] Environmental Report, H. B. Robinson Unit 2, Supplement No. 1. Answer 3.5f.

Operating Reports 3 and 4, Point Beach Nuclear Power Station.

Kahn, et al., "Radiological Surveillance Studies at a Pressurized Water Nuclear Power Reactor" op. cit.

Gland Seal Leakage

Equipment with external moving parts such as valves and the coolant pumps contain a soft packing to retard the loss of fluid and steam from the reactor system. This packing does not provide total isolation and is a major source of the coolant system leakage described previously.

In a boiling water reactor[2], a similar condition exists with regard to the turbine generator shaft. As the steam passing through the turbine was generated in the reactor vessel, it contains volatile radioactive fission and activation products such as the noble gases and iodines. In order to reduce the loss of these volatile nuclides from the turbine, process steam is bled into the outer portions of the turbine seals and removed via a gland steam condenser. The non-volatile radionuclides are condensed and the volatile radionuclides pass through a 2-minute delay line to the gaseous release discharge point. Small quantities of

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[2] A somewhat similar condition would exist in a pressurized-water reactor which is operating with appreciable primary-to-secondary leakage and fuel cladding defects.

Table 9

Estimated Gaseous Releases of Principal Radionuclides from Miscellaneous Effluents at a BWR Station

(Ci/yr per unit)

Nuclide	Turbine Gland Seal	Turbine Building	Reactor Building & Containment	Mechanical Vacuum Pump
Krypton -83m	41	10	-	-
Krypton -85m	69	16	-	-
Krypton -85	-	-	-	-
Krypton -87	200	49	-	-
Krypton -88	220	53	-	-
Krypton -89	490	17	-	-
Xenon -131m	-	-	-	-
Xenon -133m	4	1	-	-
Xenon -133	120	29	-	1445
Xenon -135m	320	82	-	-
Xenon -135	350	84	-	215
Xenon -137	900	290	-	-
Xenon -138	1,020	260	-	-
Iodine -131	0.041	0.547	0.012	-
Iodine -133	0.214	2.54	0.041	

Source: Table 3.7 of AEC Final Environmental Statement for the Duane Arnold Energy Center the steam from the seal together with the entrained volatile radionuclides also escape to the turbine building atmosphere and are released to the environment unprocessed via the turbine building exhaust. Estimates of the total discharge of volatile radionuclides from the gland seal system are shown in Table 9.

Other Sources of Leakage

Leakage from the coolant purification and waste treatment systems will be collected by the auxiliary building floor drains. Volatile radionuclides which remain entrained in the liquids will be released to the auxiliary building ventilation system and roof vents on the reactor building to the environment. In addition, gases will be released from radiochemical fume hoods during sample analysis and from tank venting and purging operations. These releases are highly variable and depend on system design parameters, construction techniques, maintenance, sampling and venting frequencies, etc. Estimates of the releases from these sources are shown in Table 6 for the PWR and Table 9 for the BWR.

During reactor startup, it is necessary to initially depressurize the cooling-water condenser. As a vacuum is drawn, the coolant present in the condenser will be partially degassed and the noble gases and a fraction of the halogens will be released. The number of startups and the intervals between shutdown and startup (which represents a decay period) are highly variable. Estimates of this frequency are about 2 -10 cold startups per year. The estimated gaseous releases for 2-3 startups per year are shown in Table 9 under the column "mechanical vacuum pump."

Atmospheric Steam Dumps from PWRs

In order to relieve high pressures in the secondary system from various abnormal operations (e.g., load rejection), PWR designs include a provision for relieving steam directly to the atmosphere through atmospheric steam dump valves. Of particular interest is the steam release which would accompany runback operations (i.e., rapid reduction of reactor power from 100% to a level at least high enough to supply the unit auxiliary load) via the main steam relief valves. The magnitude of this release can be on the order of many tens of thousands of pounds of steam in one minute. With primary-to-secondary leakage and failed fuel, radioactivity as well as steam would be released directly to the atmosphere. Although noble gases and some particulates would be released, the main concern would focus upon the release of radioiodine. Table 10 shows the estimated releases resulting from the actuation of the main steam valves for a one-minute period.

Another direct atmospheric pathway for secondary system steam exists via the feedwater heater relief valve discharge. Radioactivity release is again predicted on the concurrent presence of failed fuel and primary-to-secondary leakage. Table 10 also shows the estimated releases resulting from the actuation of the feedwater heater relief valves for a one minute period. On the order of ten thousand pounds of steam may be released during this procedure.

Table 10

PWR Radioactivity Releases via Atmospheric Steam Dumps (Ci/yr for 20 gpd Primary to Secondary Leakage and 0.2% Failed Fuel in a 3358 MWt PWR)

Curies discharged per year

Radioisotope	Main Steam Relief Valves (1 minute release)	Feedwater Heater Relief Valves (1 minute release)
Noble Gases		
Kr-85	9.06 (-5)[a]	-
Kr-88	1.30 (-4)	-
Xe-133m	1.09 (-4)	-
Xe-133	9.78 (-3)	-
Xe-135m	3.18 (-4)	-
Xe-135	2.44 (-4)	-
Radioiodine		
I-131	6.60 (-4)	1.28 (-4)
I-132	8.66 (-5)	1.68 (-5)
I-133	6.44 (-4)	1.25 (-4)
I-134	9.22 (-6)	1.78 (-6)
I-135	1.83 (-4)	3.54 (-5)
Particulates		
Mo-99	1.18 (-4)	2.28 (-5)
Tc-99m	7.32 (-5)	1.41 (-5)
Te-132	6.02 (-6)	1.16 (-6)
Cs-134	6.78 (-6)	1.31 (-7)
Cs-137	3.54 (-5)	6.64 (-6)
All Others	3.0 (-4)	4.0 (-7)
Total	1.27 (-2)	3.56 (-4)

[a] i.e., 9.06 (-5) is equivalent to 9.06×10^{-5}

Source: Trojan Nuclear Plant Final Safety Analysis Report.

DISCHARGE CONTROL OPTION CONSIDERATIONS

A variety of discharge control options were explored for both the boiling-water and pressurized-water reactor plants. The effectiveness of these options and their associated costs vary significantly. The reasons for this variation are:

(1) the differences between the two reactor types and the applicability of individual control techniques to each type of plant;

(2) the presence of multiple release pathways for the same radionuclide and lack of detailed information on the magnitude of secondary pathway releases;

(3) uncertainties in the effluent composition and chemical form;

(4) variability in the estimates of effectiveness and in the available cost data for a particular control option; and

(5) the selection and ranking of components and the order in which they are added to the baseline system.

Item 3 is particularly significant for the radioiodine releases as there is considerable uncertainty as to the chemical form of the effluent. As discussed below, differences in the chemical form of the radioiodine emissions can determine the effectiveness of control options and also greatly affect the choice of the critical exposure pathway.

Item 4 is also a major source of uncertainty in the analysis presented here as there is a scarcity of available information on system effectiveness,and inconsistencies exist in the available cost data. In particular, it is difficult to determine the cost components (operating and maintenance costs, process equipment capital costs, piping and installation costs, building and structural costs) associated with specific system cost estimates provided in the literature (1,12-15,19-24). This is especially true as much of the available utility data pertains to installing additional systems in existing plants and the additional cost for this retrofitting is not always immediately discernible. This uncertainty would lead to overestimates in the cost of installing similar systems in new plants at the design stage.

Many operating plants and those in the construction or design stages have specified treatment systems and the associated cost of these systems for attaining "lowest practicable effluent discharges" as required by the proposed Atomic Energy Commission Rulemaking, 10 CFR 50 Appendix I. This information has been extensively used (in preparing this analysis), together with information provided by the AEC (1). However, due to the number of architect-engineering firms and reactor vendors and their individual engineering and design preferences, there is a wide variety, not only in the type of control system for a particular effluent pathway, but also in the way various control options are combined in individual plant systems. This multiplicity further complicates the selection of the most cost-effective systems. In the present analysis, the attempt was made to add on components in a logical sequence. It is quite conceivable that, based on different system costs or assumptions, or actual operational experience with these systems, the order of addition of systems could differ greatly from the present analysis.

This analysis is also predicated on the occurence of certain failures and departures from optimal designed operation which conceivably may not take place during the lifetime of a particular plant. In part, these assumptions have been based on operating experience with the existing light-water reactor plants. It should be emphasized, however, that current or past experience is not sufficiently documented nor widely distributed in all cases to permit an "average" value to be adopted with confidence.

A number of parameters must be specified in order to estimate the magnitude and composition of reactor effluents. These parameters include process flow rates, leakage rates, partition factors involving various phase changes, and internal reactor system cleanup decontamination factors (DF)[3]. Tables 11 and 12 present the individual plant characteristics assumed to provide the basis for source term calculations and radioactive waste treatment system sizing.

Because of differences in design and a lack of information with respect to long term operation of the larger commercial light water reactors, no plant presently operating fits exactly the operating parameters in Tables 11 and 12. However, where possible operating experience has been factored into these estimates along with generally accepted values for various parameters (1,2,6,9,24,26-28,34,51,52).

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[3] The decontamination factor of a process is defined as:

$$DF = \frac{\text{concentration in entering stream}}{\text{concentration in effluent stream}} .$$

Table 11

BWR Plant Parameters Used in Source Term Calculations (One Unit)

Reactor power	1000 MWe		
Capacity factor	80%		
Fraction of fission products passing through:			
Condensate Demineralizer			
H-3, Y, Mo	1.0		
Cs, Rb	0.1		
Others	0.001		
Clean-up Demineralizer			
H-3, Y, Mo	1.0		
Cs, Rb	0.5		
Others	0.1	Methyl	
Partition factor (iodine in vapor/water)	Elemental	25° C	100° C
Reactor (steam/water)	0.012	0.012	0.012
Reactor Building (cold water)	0.001	1.0	
Turbine Building	1.0		1.0
Radwaste Building (hot water)	0.1		1.0
Radwaste Building (cold water)	0.001	1.0	
Gland Seal	0.1		0.1
Air Ejector	0.005	0.5	
Partition factor (other fission products in vapor/water, except H-3)	0.001		
Fraction of iodine passing through:			
Condensate Demineralizer	0.001		
Clean-up Demineralizer	0.1		
Raactor Building Filter (HEPA)	1.0		
Turbine Building Filter (HEPA)	1.0		
Radwaste Building Filter (HEPA)	1.0		
Gland Seal Condenser	0.01		
Leaks:			
Reactor Building (cold water)	1.0 gpm		
Turbine Building (steam)	2400 lb/hr		
Radwaste Building (hot water)	1 gpd		
Radwaste Building (cold water)	19 gpd		
Gland seal steam flow:	0.1% steam flow rate		

Table 12

PWR Plant Parameters Used in Source Term Calculations (One Unit)

Reactor power	1000 MWe		
Capacity factor	80%		
Number of steam generators	3		
Number of cold shutdowns per year	2		
Reactor containment volume	2.0×10^{6} cu ft		
Number of containment purges per year	4		
Blowdown rate	15 gpm		
Fraction of power from failed fuel	0.25%		
Escape rate coefficients (sec^{-1})			
Xe and Kr	6.5×10^{-8}		
I, Br, Rb, Cs	1.3×10^{-8}		
Mo	2.0×10^{-9}		
Te	1.0×10^{-9}		
Sr, Ba	1.0×10^{-11}		
Others	1.6×10^{-12}		
Fraction of fission products passing through primary coolant demineralizer (except H-3, Y, Mo, Cs, Rb)	0.1		
Cs, Rb	0.5		
H3, Y, Mo	1.0	Methyl	
Partition factor (iodine in vapor/water)	Elemental	25° C	100° C
Steam generator	0.01	0.01	0.01
Blowdown vent	0.05		1.0
Condenser air ejector	0.0005	0.5	
Containment (hot water)	0.1		1.0
Auxiliary building (hot water)	0.1		1.0
Auxiliary building (cold water)	0.0001	0.59	
Turbine building (steam)	1.0		1.0
Gland seal	0.1		0.1
Fraction of iodine passing through			
Containment filter (HEPA)	1.0		
Turbine building filter (HEPA)	1.0		
Auxiliary building filter (HEPA)	1.0		
Gland seal condenser	0.01		
Leak rate of primary coolant:			
Reactor building (hot water)	40 gpd		
Auxiliary building (hot water)	1 gpd		
Auxiliary building (cold water)	19 gpd		
Steam generator	20 gpd		
Leak rate of turbine steam	2400 lb/hr		
Gland seal steam flow	0.1% steam flow		

LIQUID DISCHARGE CONTROL OPTIONS

BWR Liquid Radwaste Systems

The liquid radioactive waste treatment system at a BWR power station is responsible for decontaminating a wide variety of waste liquids. These liquids may be divided into four general classes as shown in Table 13.

In general, four BWR liquid radioactive waste treatment systems were constructed to illustrate both the spectrum of treatment options available as well as the development that has taken place in such systems to reduce radioactivity releases. These systems are sized for a two-unit BWR station (1000 MWe per unit) and are shown in Figures 6 through 9. It should be noted that deep-bed condensate demineralizers are assumed for cleanup.

Minimum treatment is afforded by liquid radwaste system BWR-1, namely, a three day holdup of all liquid waste steams. Of the estimated 3500 Ci/year discharge from this system, about 45% originates from "clean" liquids and roughly 54% is derived from chemical wastes. The annual cost for this system is estimated at about $180,000 not including the cost of structures (Figure 6)[4].

BWR-2 (Figure 7) represents a formerly typical liquid radioactive waste treatment system in design but is sized for a two-unit (1000 MWe each) BWR power station. This system is used in BWRs now operating but many of these systems are being upgraded. Clean liquids are filtered and demineralized, allowing a 90% recycle of such liquids. Dirty liquids and laundry wastes are filtered prior to discharge. Chemical wastes are evaporated prior to discharge. Of the estimated 30 Ci/year

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[4] Annual costs include amortization and operating costs.

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Table 13

Classes of BWR Liquid Radioactive Wastes

(2 units, 1000 MWe each)

(1) Clean Liquids (reactor grade water)

- Drywell equipment drains
- Reactor building equipment drains
- Turbine building equipment drains
- Condensate demineralizer backwash
- Cleanup filter-demineralizer backwash

(2) Dirty Liquids (non-reactor grade water)

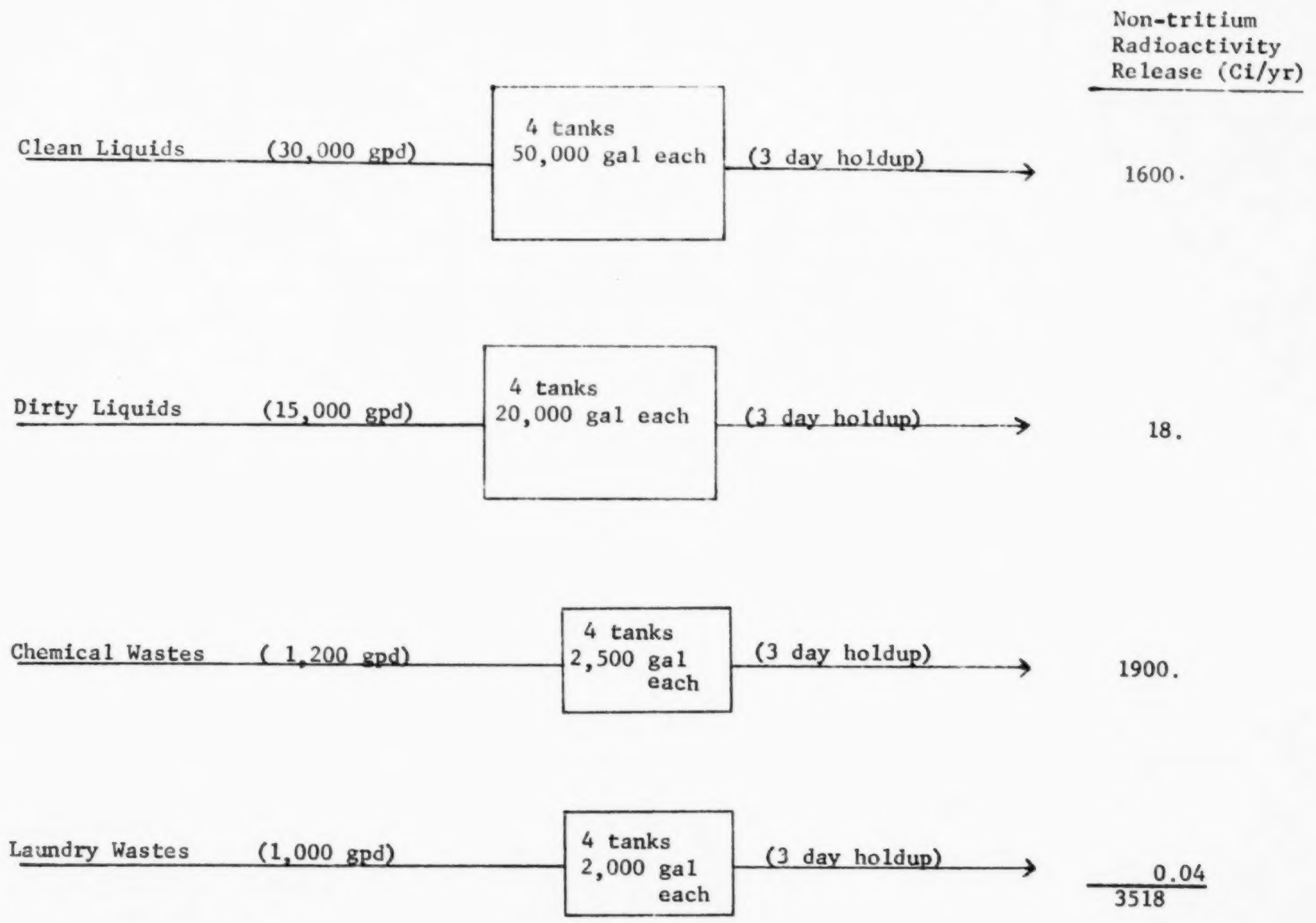
- Drywell floor drain sumps
- Reactor building floor drain sumps
- Radwaste building floor drain sumps
- Turbine building floor drain sumps

(3) Chemical Wastes

- Condensate demineralizer regeneration
- Decontamination drains
- Laboratory drains
- Shop decontamination solutions

(4) Laundry Waste

- Personnel decontamination (showers)
- Regulated shop drains
- Laundry drains
- Cask cleaning drains



Non-tritium
Radioactivity
Release (Ci/yr)
Clean Liquids
(30,000 gpd)
4 tanks
50,000 gal each
(3 day holdup)
1600.
Dirty Liquids
(15,000 gpd)
4 tanks
20,000 gal each
(3 day holdup)
18.
Chemical Wastes
(1,200 gpd)
4 tanks
2,500 gal
each
(3 day holdup)
1900.
Laundry Wastes
(1,000 gpd)
4 tanks
2,000 gal
each
(3 day holdup)
0.04
3518

Figure 6. Liquid Case BWR-1: Source Term

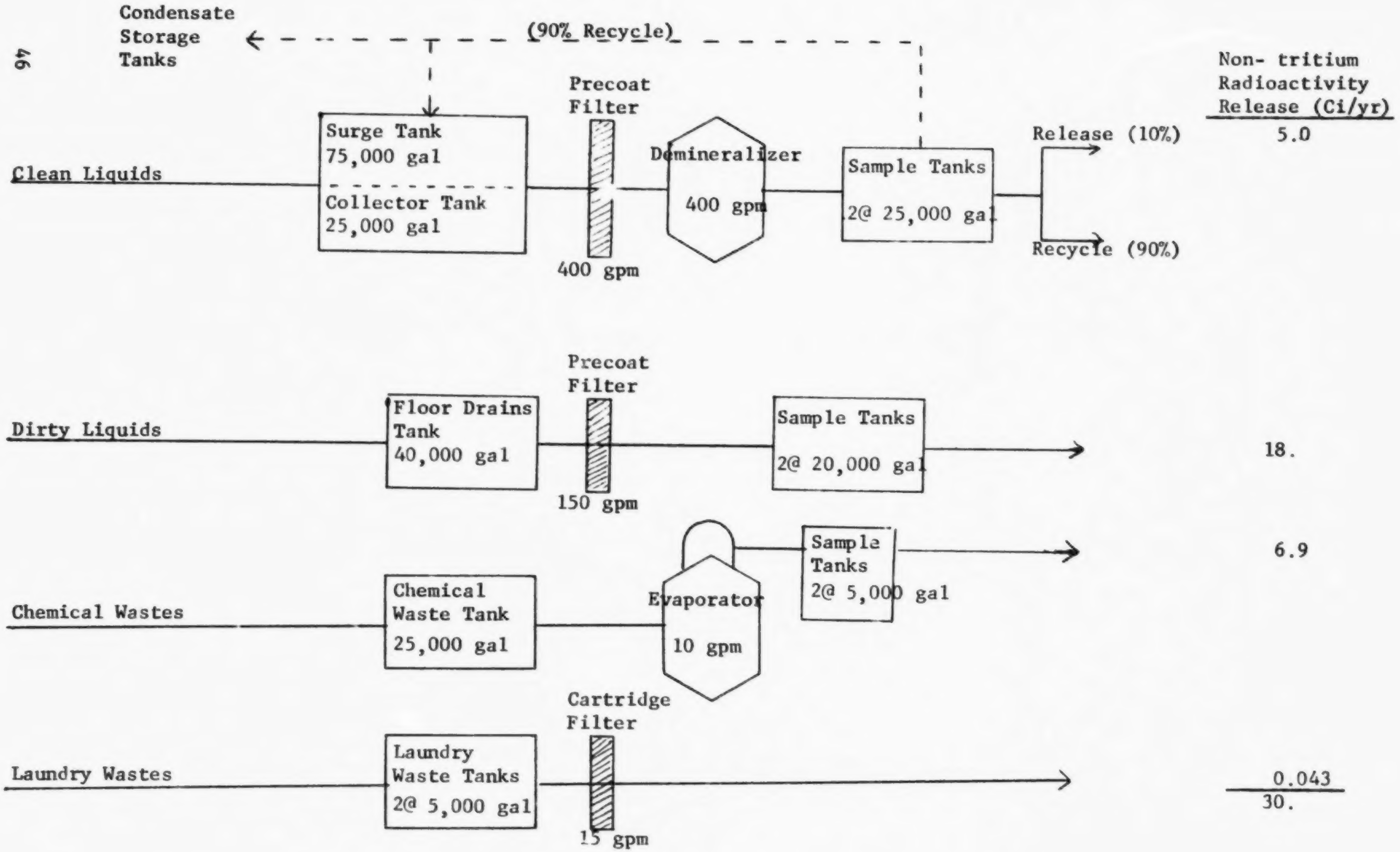

Condensate
Storage
Tanks
(90% Recycle)
Non- tritium
Radioactivity
Release (Ci/yr)
Clean Liquids
Surge Tank
75,000 gal
Collector Tank
25,000 gal
Precoat
Filter
400 gpm
Demineralizer
400 gpm
Sample Tanks
2@ 25,000 gal
Release (10%)
Recycle (90%)
5.0
Dirty Liquids
Floor Drains
Tank
40,000 gal
Precoat
Filter
150 gpm
Sample Tanks
2@ 20,000 gal
18.
Chemical Wastes
Chemical
Waste Tank
25,000 gal
Evaporator
10 gpm
Sample
Tanks
2@ 5,000 gal
6.9
Laundry Wastes
Laundry
Waste Tanks
2@ 5,000 gal
Cartridge
Filter
15 gpm
0.043
30.

Figure 7. Liquid Case BWR-2: Presently Operating

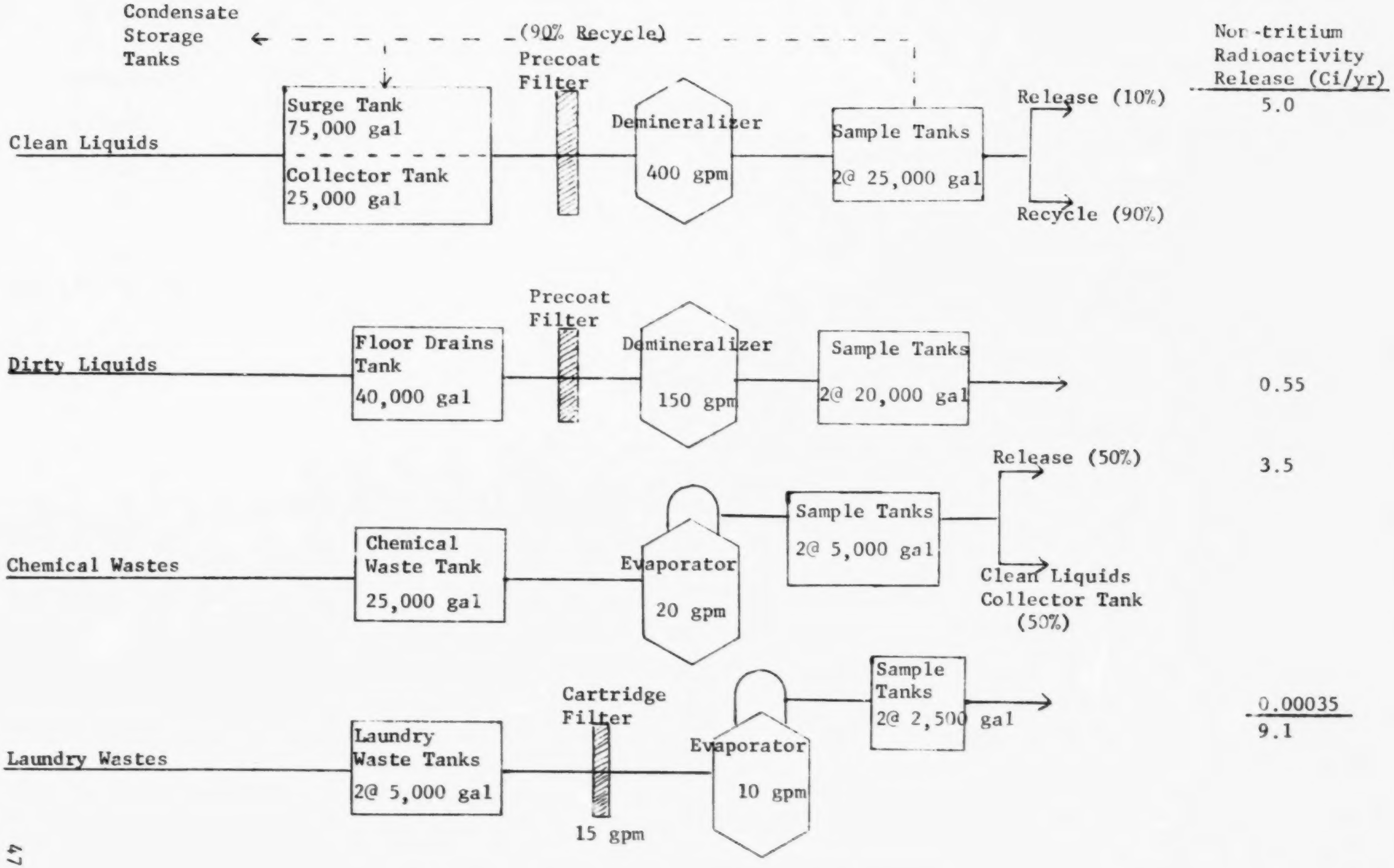

Condensate
Storage
Tanks
(90% Recycle)
Non-tritium
Radioactivity
Release (Ci/yr)
Clean Liquids
Surge Tank
75,000 gal
Collector Tank
25,000 gal
Precoat
Filter
Demineralizer
400 gpm
Sample Tanks
2@ 25,000 gal
Release (10%)
Recycle (90%)
5.0
Dirty Liquids
Floor Drains
Tank
40,000 gal
Precoat
Filter
Demineralizer
150 gpm
Sample Tanks
2@ 20,000 gal
0.55
Chemical Wastes
Chemical
Waste Tank
25,000 gal
Evaporator
20 gpm
Sample Tanks
2@ 5,000 gal
Release (50%)
Clean Liquids
Collector Tank
(50%)
3.5
Laundry Wastes
Laundry
Waste Tanks
2@ 5,000 gal
Cartridge
Filter
15 gpm
Evaporator
10 gpm
Sample
Tanks
2@ 2,500 gal
0.00035
9.1

Figure 8. Liquid Case BWR-3: Improved Design

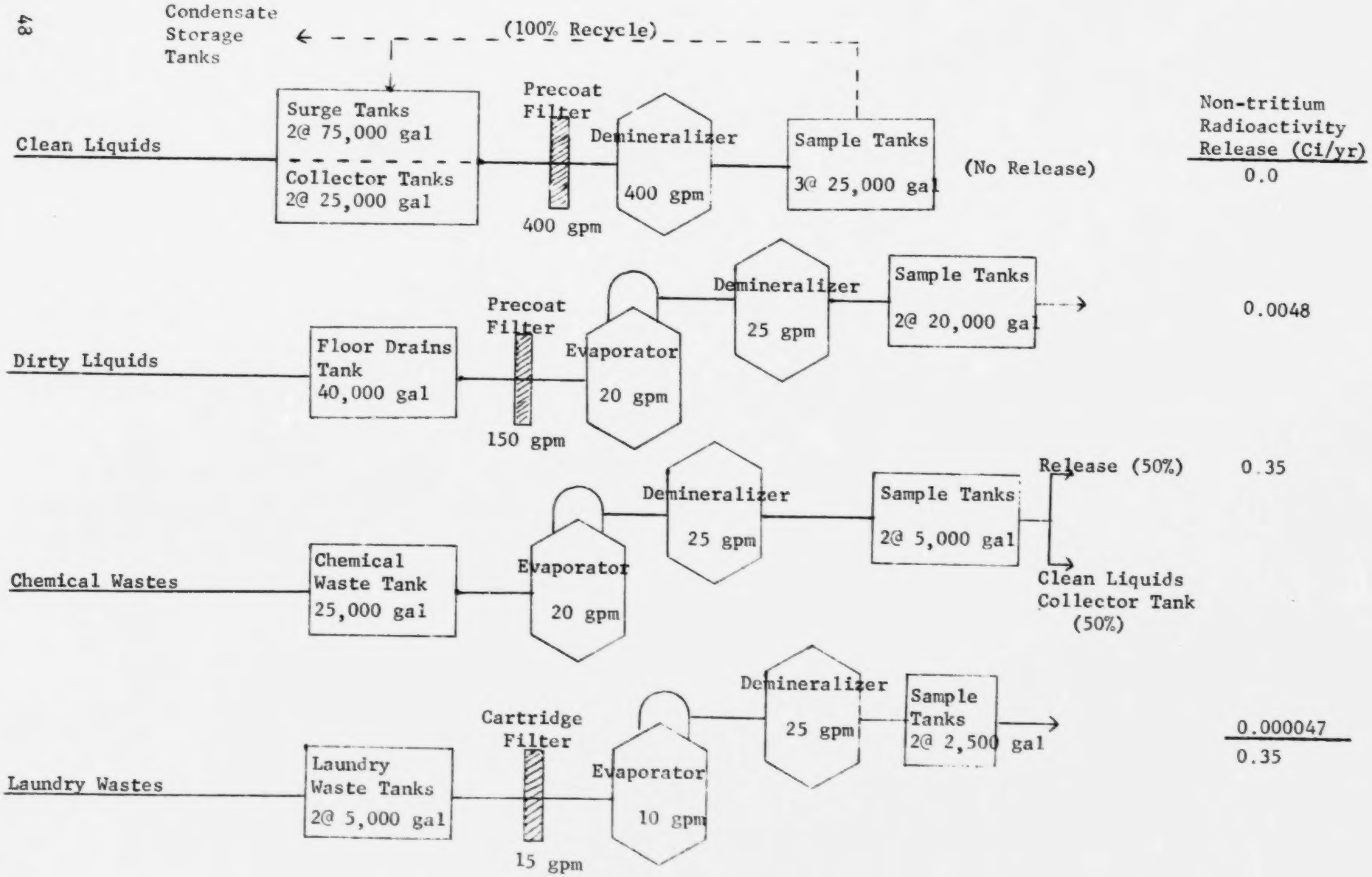

Condensate Storage Tanks
(100% Recycle)
Clean Liquids
Surge Tanks 2@ 75,000 gal
Collector Tanks 2@ 25,000 gal
Precoat Filter
400 gpm
Demineralizer
400 gpm
Sample Tanks 3@ 25,000 gal
(No Release)
Non-tritium Radioactivity Release (Ci/yr)
0.0
Dirty Liquids
Floor Drains Tank 40,000 gal
Precoat Filter
150 gpm
Evaporator
20 gpm
Demineralizer
25 gpm
Sample Tanks 2@ 20,000 gal
0.0048
Chemical Wastes
Chemical Waste Tank 25,000 gal
Evaporator
20 gpm
Demineralizer
25 gpm
Sample Tanks 2@ 5,000 gal
Release (50%)
Clean Liquids Collector Tank (50%)
0.35
Laundry Wastes
Laundry Waste Tanks 2@ 5,000 gal
Cartridge Filter
15 gpm
Evaporator
10 gpm
Demineralizer
25 gpm
Sample Tanks 2@ 2,500 gal
0.000047
0.35

Figure 9. Liquid Case BWR-4: Maximum Treatment

release, almost 61% comes from the release of dirty liquids. Annual cost for this system, not including structures, is estimated at $401,000.

An example of an improved BWR liquid radwaste system is shown as system BWR-3, Figure 8. This technology is planned for plants now being built. Clean and dirty liquids are filtered and demineralized. For the purposes of determining radioactivity discharges it is assumed that 10% of the clean liquids and 100% of the dirty liquids are discharged, although it may be possible to recycle a portion of the treated dirty liquids. Chemical wastes and laundry wastes are evaporated but 50% of the chemical wastes are recycled after treatment. Of the estimated 9.1 Ci/year release, 55% originates from clean liquids and almost 39% from chemical wastes. Annual cost for this system without structures is estimated at $560,000.

"Maximum" treatment is afforded each waste stream in system BWR-4, Figure 9. Clean liquids are filtered and demineralized but additional tankage is added to assure complete recycling of these liquids. Dirty liquids, chemical wastes, and laundry wastes are evaporated and demineralized; 50% of the chemical wastes are recycled. Virtually all of the estimated 0.35 Ci/year release originates from chemical wastes. Estimated annual cost for this system is $788,000.

In order to calculate radioactivity releases, the plant parameters detailed in Table 11 were utilized to derive the source term, system BWR-1 in Table 16. Table 14 was constructed to show the total decontamination factor (DF) accorded each waste stream in each system. Individual component DFs, which make up these total DFs, are shown in

Table 14

DFs for BWR Liquid Systems

Treatment System	Total DF				
	I	Cs,Rb	Mo[a]	Y[a]	Others
BWR-1					
Clean	1	1	1	1	1
Dirty	1	1	1	1	1
Chemical	1	1	1	1	1
Laundry	1	1	1	1	1
BWR-2					
Clean	10^2	10	10^2	10	10^2
Dirty	1	1	1	1	1
Chemical	10^2	10^4	10^6	10^5	10^3
Laundry	1	1	1	1	1
BWR-3					
Clean	10^2	10	10^2	10	10^2
Dirty	10^2	10	10^2	10	10^2
Chemical	10^2	10^4	10^6	10^5	10^3
Laundry	10^2	10^2	10^4	10^3	10^2
BWR-4					
Clean	← Recycled →				
Dirty	10^3	10^5	10^6	10^5	10^4
Chemical	10^3	10^5	10^6	10^5	10^4
Laundry	10^3	10^3	10^4	10^3	10^3

[a] Includes an additional DF of 10^2 for Mo and 10 for Y to account for plateout, filtration, and demineralization, where applicable.

Table 15

Liquid Radwaste System Component DFs

Components	Decontamination Factors (DFs)				
	I	Cs,Rb	Y	Mo	Others
DEMINERALIZERS					
- PWR -					
Mixed Bed (Li_3BO_3 form, CVCS)(1,2)	10	2	1	1	10
Mixed Bed (steam generator blowdown)(3,6,9)	10	2	1	1	50
- BWR -					
Mixed Bed (H-OH form, clean waste)(1,11)	10^2	10	1	1	10^2
- PWR & BWR -					
Mixed bed in evaporator condensate(1,6)	10	10	1	1	10
EVAPORATORS (PWR & BWR)					
Waste(5,7)	10^2	10^4	10^4	10^4	10^3
Laundry(1)	10^2	10^2	10^2	10^2	10^2
OTHER					
Removal of Mo and Y by plating out, filtration, demineralization, etc.(1)	-	-	10	10^2	-

Table 16

Releases of Long-Lived Radionuclides for BWR Liquid Radwaste Systems

Annual Release[a] (Ci/yr)

Radionuclide	Half-Life	Treatment Option BWR-1	 BWR-2	 BWR-3	 BWR-4
Mn-54	303.0 d	3.7(0)	6.2(-1)	8.1(-3)	1.9(-4)
Fe-55	2.6 y	1.8(2)	4.9(-1)	1.1(-1)	7.5(-3)
Fe-59	45.0 d	3.8(0)	1.7(-2)	2.6(-3)	1.3(-4)
Co-58	71.3 d	2.9(2)	1.0(0)	1.8(-1)	1.1(-2)
Co-60	5.26y	4.8(1)	1.3(-1)	2.8(-2)	2.1(-3)
Sr-89	52.0 d	4.4(2)	1.4(0)	2.8(-1)	1.7(-2)
Sr-90	28.1 y	9.9(0)	1.0(-1)	2.7(-2)	2.1(-3)
Y-91	58.8 d	2.0(2)	4.2(0)	2.2(0)	2.9(-5)
Zr-95	65.0 d	5.2(0)	1.6(-2)	3.2(-3)	2.1(-4)
Nb-95	35.0 d	6.5(0)	1.7(-2)	3.8(-3)	2.8(-4)
Ru-103	39.6 d	2.7(0)	1.0(-2)	1.8(-3)	1.0(-4)
Ru-106	367.0 d	1.6(0)	3.9(-3)	9.3(-4)	7.0(-5)
Cs-134	2.05y	7.4(1)	2.0(-1)	1.0(-1)	3.3(-4)
Cs-137	30.0 y	6.3(1)	1.6(-1)	8.5(-2)	2.8(-4)
Ce-141	33.0 d	9.1(0)	4.5(-2)	6.7(-3)	2.8(-4)
Ce-144	284.0 d	4.5(0)	1.1(-2)	2.7(-3)	2.0(-4)
I-131	8.0 d	5.9(2)	6.0	2.7(0)	2.6(-1)
I-133	21.0 h	2.1(2)	2.4	9.8(-2)	9.7(-4)
Total non-tritium releases		3500.0	30.0	9.1	0.35
Total Tritium[b]		200.0	200.0	200.0	130.0

[a] For a two unit, 1000 MWe each, BWR Power Station. Releases are written in exponential notation, i.e., 8.8(-1) = 8.8 x 10^{-1} = 0.88.

[b] A conservative estimate based partly on operating experience

Table 15. After taking into account the total DFs afforded each waste stream, and allowing appropriate credit for recycle where applicable, total radioactivity releases were determined for each of the four systems. Table 16 shows the estimated releases by system for the longer-lived or more significant radionuclides as well as total radioactivity releases for each system. Tritium releases were estimated to be about 200 Ci/year for all systems except the maximum treatment system, where a complete recycle of clean liquids reduces tritium discharges to about 130 Ci/year.

PWR Liquid Radwaste Systems

The liquid radioactive waste treatment system in a PWR power station is also responsible for decontaminating a wide variety of waste liquids. These liquids may be divided into five general classes, each of which has one or more components as shown in Table 17.

In general, four PWR liquid radioactive waste treatment systems were constructed to illustrate both the spectrum of treatment options available as well as the development that has taken place in such systems to reduce radioactivity releases. These systems are sized for a two unit (1000 MWe per unit) PWR power station and are shown in Figures 10 through 13.

Minimum treatment is provided by liquid radwaste system PWR-1, Figure 10. Clean and dirty liquids are released after a three day holdup, while laundry wastes are afforded a 30-day holdup prior to release. Steam generator blowdown and turbine building drains liquids are released without any holdup. Of the estimated 3600 Ci/year discharge, about 70 percent originates from clean liquids and nearly 30% comes

Table 17

Classes of PWR Liquid Radioactive Wastes

(2 units, 1000 MWe each)

(1) Clean Liquids (reactor grade water)

Reactor coolant pump seal leakage
Equipment leakage
Valve leakoffs
Reactor vessel flange leakoffs
Resin flush
Filter changes
Heat exchanger, pump, and tank maintenance
CVCS letdown

(2) Dirty Liquids (non-reactor grade water)

Auxiliary building floor drains
Equipment leakage
Containment sump
Fuel building sump
Chemical laboratory drains
Decontamination area drains

(3) Steam Generator Blowdown

Steam generator blowdown

(4) Turbine Building Drains

Secondary system leakage to turbine building drains

(5) Laundry Wastes

Hot shower drains
Laundry

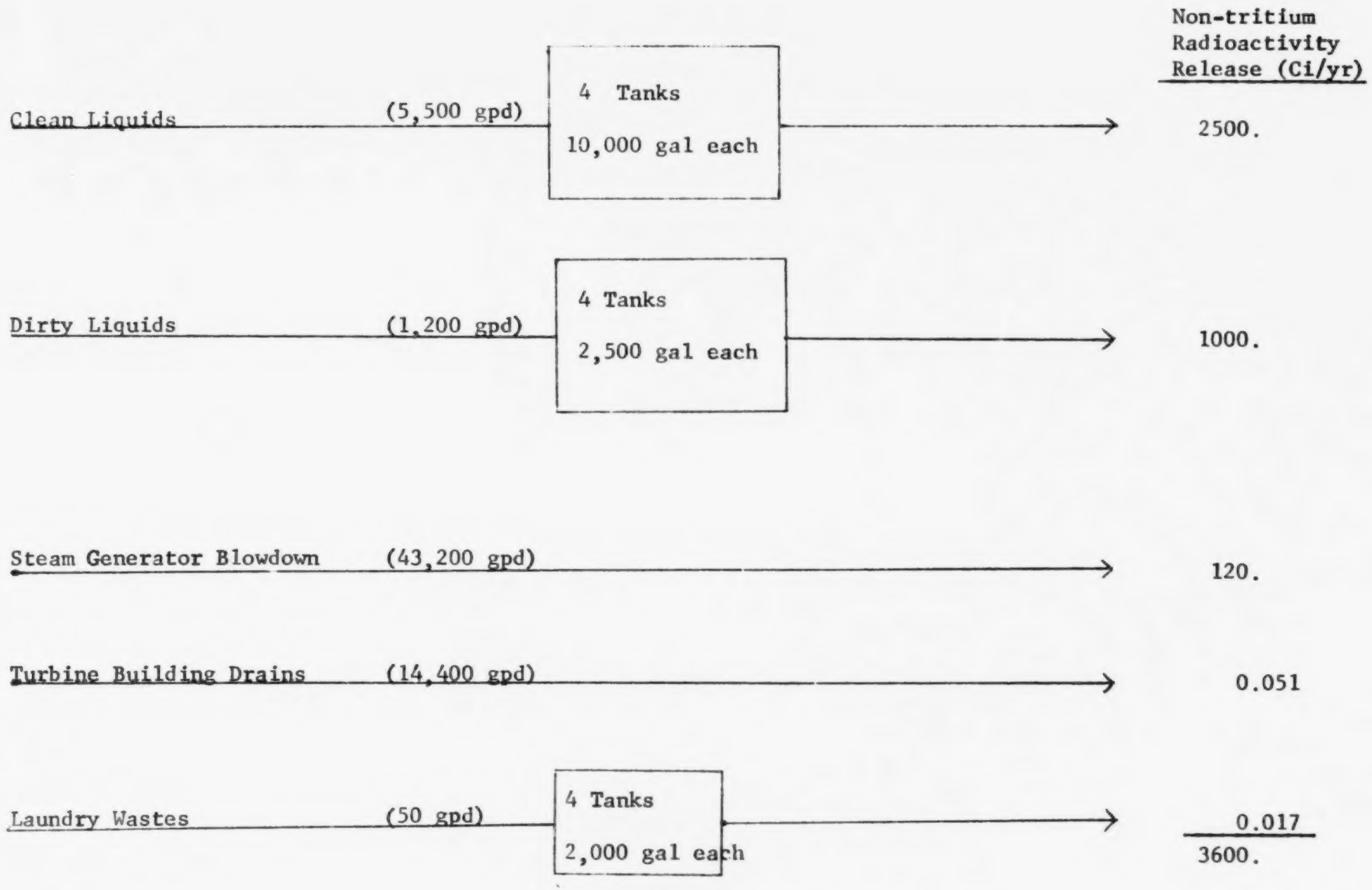
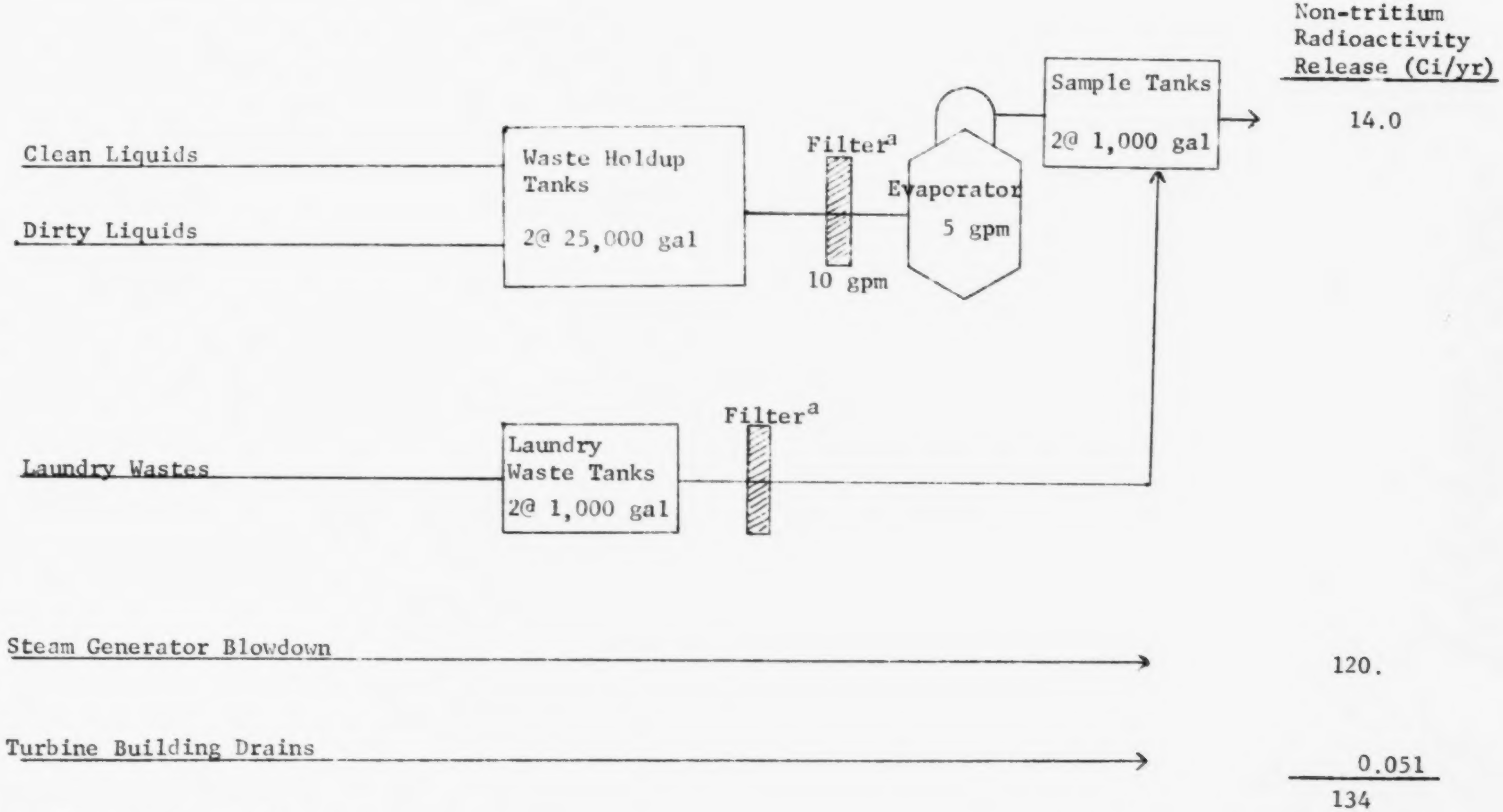
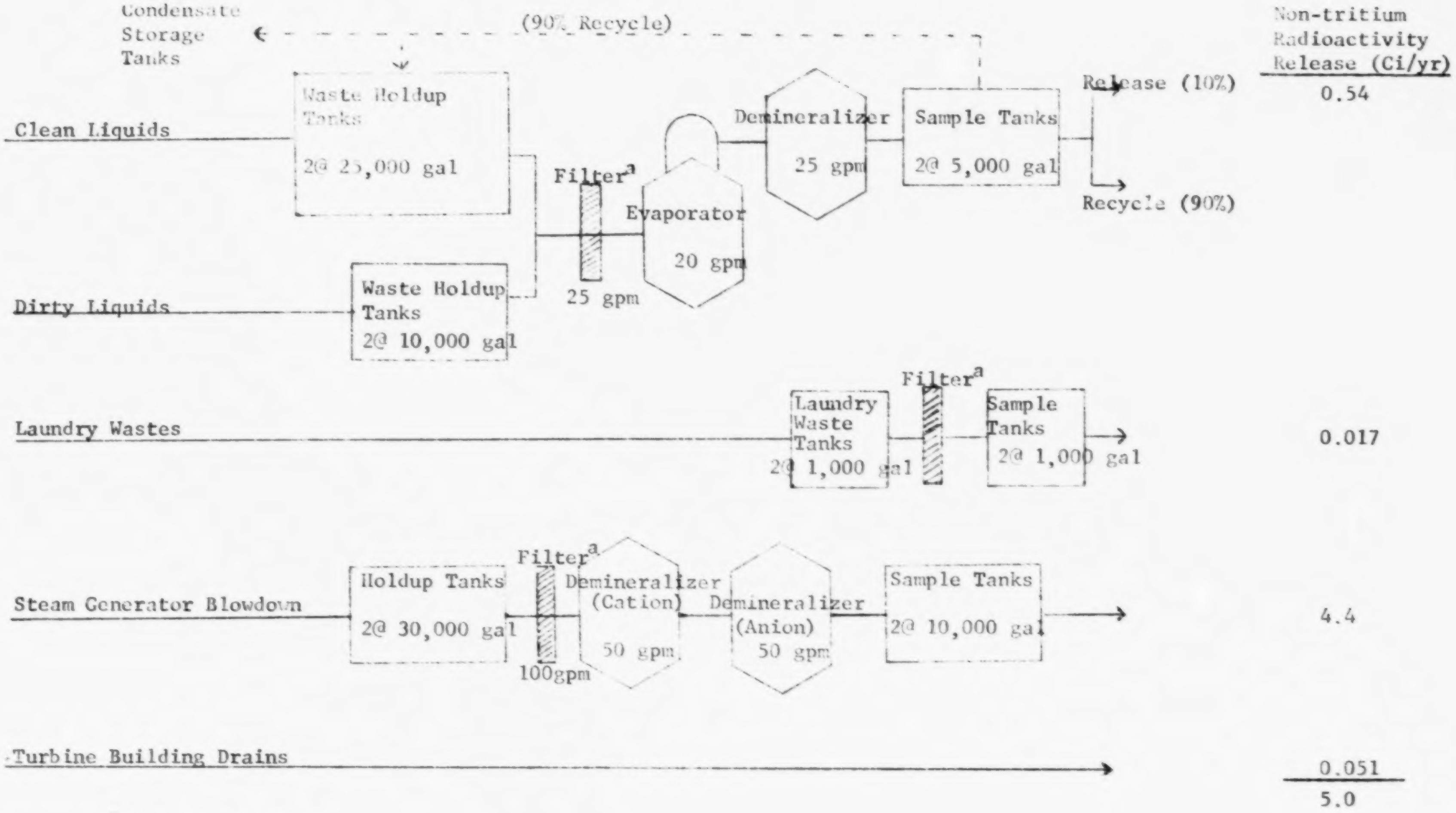

Non-tritium
Radioactivity
Release (Ci/yr)
Clean Liquids
(5,500 gpd)
4 Tanks
10,000 gal each
2500.
Dirty Liquids
(1,200 gpd)
4 Tanks
2,500 gal each
1000.
Steam Generator Blowdown
(43,200 gpd)
120.
Turbine Building Drains
(14,400 gpd)
0.051
Laundry Wastes
(50 gpd)
4 Tanks
2,000 gal each
0.017
3600.

Figure 10. Liquid Case PWR-1: Source Term


Non-tritium
Radioactivity
Release (Ci/yr)
Clean Liquids
Dirty Liquids
Waste Holdup
Tanks
2@ 25,000 gal
Filter[a]
10 gpm
Evaporator
5 gpm
Sample Tanks
2@ 1,000 gal
14.0
Laundry Wastes
Laundry
Waste Tanks
2@ 1,000 gal
Filter[a]
Steam Generator Blowdown
120.
Turbine Building Drains
0.051
134

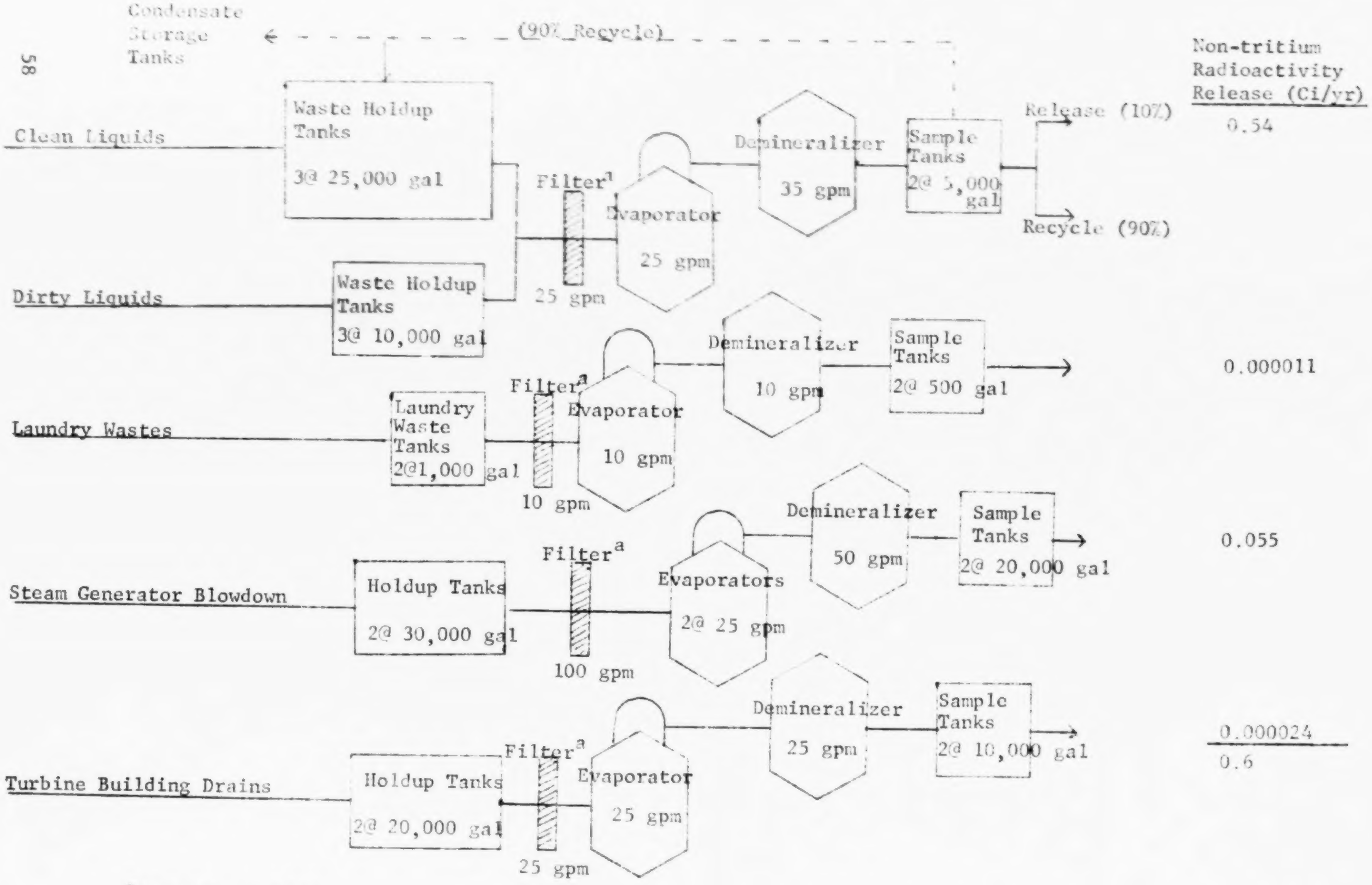
[a] Cartridge Filter

Figure 11. Liquid Case PWR-2: Presently Operating


Condensate
Storage
Tanks
(90% Recycle)
Clean Liquids
Waste Holdup
Tanks
2@ 25,000 gal
Dirty Liquids
Waste Holdup
Tanks
2@ 10,000 gal
Filter[a]
25 gpm
Evaporator
20 gpm
Demineralizer
25 gpm
Sample Tanks
2@ 5,000 gal
Release (10%)
Recycle (90%)
Non-tritium
Radioactivity
Release (Ci/yr)
0.54
Laundry Wastes
Laundry
Waste
Tanks
2@ 1,000 gal
Filter[a]
Sample
Tanks
2@ 1,000 gal
0.017
Steam Generator Blowdown
Holdup Tanks
2@ 30,000 gal
Filter[a]
100gpm
Demineralizer
(Cation)
50 gpm
Demineralizer
(Anion)
50 gpm
Sample Tanks
2@ 10,000 gal
4.4
Turbine Building Drains
0.051
5.0

[a]Cartridge Filter

Figure 12. Liquid Case PWR-3: Improved Design


Condensate Storage Tanks
(90% Recycle)
Non-tritium Radioactivity Release (Ci/yr)
Clean Liquids
Waste Holdup Tanks 3@ 25,000 gal
Filter[a]
25 gpm
Evaporator 25 gpm
Demineralizer 35 gpm
Sample Tanks 2@ 5,000 gal
Release (10%)
Recycle (90%)
0.54
Dirty Liquids
Waste Holdup Tanks 3@ 10,000 gal
Laundry Wastes
Laundry Waste Tanks 2@1,000 gal
Filter[a]
10 gpm
Evaporator 10 gpm
Demineralizer 10 gpm
Sample Tanks 2@ 500 gal
0.000011
Steam Generator Blowdown
Holdup Tanks 2@ 30,000 gal
Filter[a]
100 gpm
Evaporators 2@ 25 gpm
Demineralizer 50 gpm
Sample Tanks 2@ 20,000 gal
0.055
Turbine Building Drains
Holdup Tanks 2@ 20,000 gal
Filter[a]
25 gpm
Evaporator 25 gpm
Demineralizer 25 gpm
Sample Tanks 2@ 10,000 gal
0.000024
0.6

[a] Cartridge Filter

Figure 13. Liquid Case PWR-4: Maximum Treatment

from dirty liquids. Annual cost for this system is estimated at $52,000 without structures.

System PWR-2 (shown in Figure 11) is typical of many PWRs now operating. Clean and dirty liquids are evaporated prior to release, while laundry wastes are only filtered and released. Steam generator blowdown and turbine building drains are released without treatment. Of the estimated 134 Ci/year which are discharged, almost 90% may be attributed to steam generator blowdown. Estimated annual cost for this system is $121,000.

An example of an improved system presently planned for PWRs being built is shown as system PWR-3 in Figure 12. Clean and dirty liquids are evaporated and demineralized, allowing a 90% recycle of these liquids. Steam generator blowdown is passed through two demineralizers in series prior to release. Laundry wastes are filtered and discharged while turbine building drains liquids are released untreated. The estimated annual release is reduced to about 5 Ci/year, of which almost 90% is derived from steam generator blowdown. Annual cost for this system, without structures, is about $280,000.

"Maximum" treatment of each waste stream is provided in system PWR-4, Figure 13. All waste streams are evaporated and demineralized. Extra tankage and processing capability is added to assure 90% recycle capability for clean and dirty liquids and to lower radioactivity concentrations of recycled liquids. All other effluent streams are released after treatment. Total release for this "maximum" treatment alternative is estimated at 0.60 Ci/year of which 90% is derived from the discharge of

Table 18

DFs for PWR Liquid Systems

Treatment System	Total DF[a]				
	I	Cs,Rb	Mo[b]	Y[b]	Others
PWR-1					
Clean	1	1	1	1	1
Dirty	1	1	1	1	1
S. G. Blowdown	1	1	1	1	1
Turbine Bldg	1	1	1	1	1
Laundry	1	1	1	1	1
PWR-2					
Clean	10^2	10^4	10^6	10^5	10^3
Dirty	10^2	10^4	10^6	10^5	10^3
Laundry	1	1	1	1	1
S. G. Blowdown	1	1	1	1	1
Turbine Bldg	1	1	1	1	1
PWR-3					
Clean	10^3	10^5	10^6	10^5	10^4
Dirty	10^3	10^5	10^6	10^5	10^4
Laundry	1	1	1	1	1
S. G. Blowdown	10^2	4	10^2	10	2.5×10^2
Turbine Bldg	1	1	1	1	1
PWR-4					
Clean	10^3	10^5	10^6	10^5	10^4
Dirty	10^3	10^5	10^6	10^5	10^4
Laundry	10^3	10^3	10^4	10^3	10^3
S. G. Blowdown	10^3	10^5	10^6	10^5	10^4
Turbine Bldg	10^3	10^5	10^6	10^5	10^4

[a] Excludes DF exerted by CVCS system on liquids eventually discharged.

[b] Includes an additional DF of 10^2 for Mo and 10 for Y to account for plateout, filtration, and demineralization, where applicable.

Table 19

Releases of Long-Lived Radionuclides For PWR Liquid Radwaste Systems

Annual Release[a] (Ci/yr)

Radionuclide	Half-Life	Treatment of Option			
		PWR-1	PWR-2	PWR-3	PWR-4
Mn-54	303.0 d	8.8(-1)	3.3(-2)	1.6(-4)	1.2(-5)
Fe-55	2.6 y	2.5(0)	9.1(-2)	4.4(-4)	3.3(-5)
Fe-59	45.0 d	7.2(-1)	2.7(-2)	1.2(-4)	9.5(-6)
Co-58	71.3 d	2.4(1)	9.1(-1)	3.6(-1)	3.2(-4)
Co-60	5.26y	2.5(0)	9.1(-2)	3.6(-2)	1.8(-6)
Sr-89	52.0 d	1.3(0)	6.1(-2)	2.8(-4)	1.9(-5)
Sr-90	28.1 y	4.4(-2)	2.0(-3)	9.5(-6)	6.2(-7)
Y-91	58.8 d	9.7(0)	4.8(-1)	4.8(-2)	1.4(-5)
Zr-95	65.0 d	2.2(-1)	9.9(-3)	4.6(-5)	3.1(-6)
Nb-95	35.0 d	2.1(-1)	9.9(-3)	4.8(-5)	3.0(-6)
Ru-103	39.6 d	1.6(-1)	7.1(-3)	3.3(-5)	2.3(-6)
Ru-106	367.0 d	4.2(-2)	1.8(-3)	8.6(-6)	5.8(-7)
Cs-134	2.05y	2.9(2)	4.6(0)	1.2(0)	3.4(-4)
Cs-137	30.0 y	2.7(2)	4.0(0)	1.0(0)	3.1(-4)
Ce-141	33.0 d	2.4(-1)	1.0(-2)	4.7(-5)	3.3(-6)
Ce-144	284.0 d	1.4(-1)	6.2(-3)	2.9(-5)	1.9(-6)
I-131	8.0 d	8.3(2)	4.1(1)	4.3(-1)	1.1(-1)
I-133	21 h	4.4(2)	1.8(1)	1.9(-1)	5.7(-2)
Total non-tritium releases		3600.0	134.0	5.0	0.60
Total Tritium[b]		1200.0	1200.0	760.0	760.0

---

[a] For a two unit, 1000 MWe each, PWR Power Station. Releases are written in exponential notation, i.e., 8.8(-1) = 8.8 x 10^{-1}.

[b] Based on operating experience of smaller plants

treated clean and dirty liquids. Estimated annual cost for this system is $879,000 with structures.

In order to calculate radioactivity releases, the plant parameters detailed in Table 12 were utilized to derive the source term, system PWR-1 in Table 19. With this source term, Table 18 was constructed to show the total decontamination factor (DF) accorded each waste stream in each radwaste system. Individual component DFs, which make up these system DFs, are shown in Table 15. After taking into account the total DFs afforded each waste stream in each system, and allowing appropriate credit for recycle where applicable, total radioactivity releases were determined for each of the four systems. Table 19 lists these radioactivity releases by system for the longer-lived radionuclides as well as total radioactivity releases for each system. Tritium releases are estimated to be about 1,200 Ci/yr for the two-unit (1000 MWe each) PWR power station using liquid radwaste systems PWR-1 or PWR-2. A tritium release of 760 Ci/yr is estimated for the same PWR power station using liquid radwaste systems PWR-3 or PWR-4.

Cost Analysis

Having selected four PWR and four BWR liquid radwaste systems, the major components of each were broken down as shown in Tables 20 and 21. In order to determine annual costs for each system, a fixed charge rate of 16.6% of the capital investment (without structures) was added to operating and maintenance costs (1). Capital costs were estimated from reference (1) and supplemented by estimates of the nuclear industry (13, 14, 15, 19-24). Aside from being scarce, available estimates of operating and maintenance costs are quite variable for similar equipment

Table 20A

Equipment, Annual, and Capital Costs^{a} (BWR)

Equipment Items	BWR-1	BWR-2	BWR-3	BWR-4
	(Number of Equipment Items Required for Given Systems)			
Tankage (gal)				
2,000	4			
2,500	4		2	2
5,000		4	4	4
20,000	4	2	2	2
25,000		4	4	6
40,000		1	1	1
50,000	4			
75,000		1	1	2
Filters (Precoat) (gpm)				
150		1	1	1
400		1	1	1
Filter (Cartridge) (gpm)				
15		1	1	1
Demineralizers (gpm)				
25				3
150			1	
400		1	1	1
Evaporators (gpm)				
10		1	1	1
20			1	2
Estimated Capital Cost	$918,000	$1,738,000	$2,344,000	$3,231,000
Estimated Annual Cost	$180,000	$ 401,000	$ 560,000	$ 788,000

---

[a] Without structures.

Table 20B

Equipment, Annual, and Capital Costs[a] (PWR)

	PWR-1	PWR-2	PWR-3	PWR-4
Equipment Items	(Number of Equipment Items Required for Given Systems)			
Tankage (gal)				
500				2
1,000		4	2	2
2,000	4			
2,500	4			
5,000			2	2
10,000	4		2	5
20,000			2	4
25,000		2	2	3
30,000			2	2
Filters (cartridge, gpm)				
10		2	1	1
25			1	2
100			1	1
Demineralizers (gpm)				
10				1
25			1	1
35				1
50			2	1
Evaporators (gpm)				
5		1		1
20			1	
25				4
Estimated Capital Cost	$264,000	$509,000	$1,213,000	$3,547,500
Estimated Annual Cost	$ 52,000	$121,000	$ 280,000	$ 879,000

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[a] Without structures.

Table 21

LIQUID RADIOACTIVE WASTE SUMMARY TABLE: BWRs AND PWRs

Discharge Control Option	System Designation	Estimated Capital Cost	Estimated Annual Cost	LIQUID NON-TRITIUM RADIOACTIVITY RELEASE (Ci/yr)							TRITIUM (Ci/yr)
				Clean	Dirty	Chemical Waste	Laundry Waste	Steam Generator Blowdown	Turbine Drains	TOTAL	
Pressurized Water Reactors (d)											
Source Term	PWR-1	$ 264,000	$ 52,000	2500.	1000.		0.017	120.	0.051	3600	1200
Presently Operating	PWR-2	$ 509,000	$ 121,000	14.	(a)		(a)	120.	0.051	134	1200
Improved	PWR-3	$1,213,000	$ 280,000	0.54	(a)		0.017	4.4	0.051	5.0	760
Maximum Treatment	PWR-4	$3,547,000	$ 879,000	0.54	(a)		(b)	0.055	(b)	0.6	760
Boiling Water Reactors (d)											
Source Term	BWR-1	$ 918,000	$ 180,000	1600.	18.	1900.	0.04			3518	200
Presently Operating	BWR-2	$1,739,000	$ 401,000	5.	18.	6.9	0.04			30	200
Improved	BWR-3	$2,344,000	$ 560,000	5.	0.55	3.5	(b)			9.1	200
Maximum Treatment	BWR-4	$3,231,000	$ 788,000	(c)	0.0048	0.35	(b)			0.35	130

(a) Included with clean liquids
(b) Less than 10^{-3} Ci/yr
(c) No release, 100% recycled
(d) Values are for two units, 1000 MWe each.

items (12,13,14,15). Therefore, operating and maintenance costs were estimated at 3% of the capital cost for tankage and 10% of the capital cost for filters, evaporators, and demineralizers. Table 21 relates the non-tritium activity release to annual costs for BWR and PWR liquid radwaste systems.

Estimated costs are dependent upon the specific systems chosen. These systems in turn are related to the mode of plant operation assumed. For example, it was assumed above that the BWR operated with deep bed condensate demineralizers. Also available are Powdex filter-demineralizers. Although the on-line operating costs of the Powdex units appear less than those of the deep bed condensate demineralizers, operating experience with the Powdex units in BWRs is limited and the increased cost of shipping the spent resins (Powdex units are not regenerated) tends to increase total operating costs. In PWRs, different reactor vendors handle secondary system cleanup in different ways. Westinghouse prefers to blowdown the steam generators at a controlled rate whereas Babcock and Wilcox employs full-flow condensate demineralizers (3, 12, 16). Although B & W plants avoid a continuous blowdown stream from the steam generators, the solution eluted from regenerating the condensate demineralizers must be treated when contaminated by primary-to-secondary leakage. North Anna 3 and 4 will employ a 25 gpm evaporator for this purpose (14). Combustion Engineering employs both steam generator blowdown (though at a lower flow rate than Westinghouse plants) and partial-flow condensate demineralizers (7, 17). Liquids from blowdown and condensate demineralizer regeneration may be handled by a liquid radioactive waste disposal system of slightly larger capacity. The use of condensate demineralizers in B & W and C-E PWRs necessitates cleanup by evaporation upon regeneration after operating with primary-to-secondary leakage. This increases the annual costs of secondary system cleanup by almost a factor of three but would increase total liquid radwaste system annual costs for PWR-3, Table 21, by only about 30%.

As shown by the selection of liquid radwaste systems, no redundancy of critical radwaste system components is assumed. However, some inherent redundancy does exist in systems presently planned (PWR-3 and BWR-3) for BWRs and PWRs. Recent operating reports for a few Westinghouse PWRs show, for example, that boric acid evaporators (of the Chemical and Volume Control System) have been used to treat steam generator blowdown although not necessarily a desirable alternative. PWRs of Babcock and Wilcox design or Combustion Engineering design typically provide evaporators which may be used interchangeably for boron recovery or miscellaneous liquid waste processing (3, 7, 16, 17). BWRs with only two evaporators (laundry waste and chemical waste) and only two demineralizers (clean liquids and dirty liquids) could share equipment between these respective systems to achieve lowest practicable releases. Sizings indicated in system BWR-3, Figure 21, would allow this sharing until repairs on the defective evaporator and/or demineralizer could be completed. Therefore, it appears that sufficient flexibility exists in BWR and PWR liquid radwaste systems presently planned to maintain radioactivity releases and dose equivalents very close to estimated levels.

Finally, estimated costs for liquid systems do not include structures and the costs of appropriate solid waste systems. Consideration has been given to the fact that the capital cost of structures may be on the order of a few million dollars and that solid waste system capacity must be increased with more sophisticated liquid radwaste systems, resulting in perhaps one-half million dollars in capital cost (without structures) and $200,000 - $500,000 in annual cost (1). On the other hand, building space and solid waste systems are already planned for integration with liquid radwaste systems as sophisticated as BWR-3 and PWR-3, respectively. Although these items do not directly reduce radioactivity in liquid effluents, their respective costs must also be considered.

NOBLE GAS DISCHARGE CONTROL OPTIONS

Pressurized Water Reactors (PWRs)

Radioactive isotopes of krypton and xenon, the noble gases of greatest concern, are generated inside the fuel rods of pressurized water reactors. These gaseous radionuclides escape through fuel cladding defects and enter the primary reactor coolant system. Due to leakages of primary coolant, intentional or otherwise, the radioactive isotopes of krypton and xenon may be released to the environment. These release pathways may be broken down as follows:

(1) Primary Gases
    - Shim Bleed
    - Shutdown Degasification

(2) Secondary System Gases
    - Air Ejector Exhaust
    - Gland Seal Exhaust
    - Steam Generator Blowdown Tank (SGBT) Vent

(3) Building Ventilation
    - Containment Purge
    - Auxiliary Building
    - Turbine Building

Whereas the primary gas sources (shim bleed and shutdown degasification may be routed to the waste gas treatment system, those gases resulting from primary coolant leakages (secondary system gases and building ventilation) usually escape to the atmosphere untreated.

In order to determine the effectiveness of various PWR noble gas discharge control options, the source term for each of the release pathways detailed above was calculated. Assumptions for calculating

Table 22

PWR Noble Gas Source Term: 2 Units, 1,000 MWe Each (Ci/yr)

Nuclide	Half-life (31)	Primary System Gases	Secondary System Gases			Building Ventilation			Total
			Blowdown Vent	Air Ejector Exhaust	Gland Seal Exhaust	Containment Purge	Auxiliary Building	Turbine Building	
Kr-83m	1.86h	4.2(2)[a]	--[b]	3.0(0)	--	--	3.0(0)	--	4.2(2)
Kr-85m	4.4h	2.3(3)	--	1.6(1)	--	--	1.6(1)	--	2.3(3)
Kr-85	10.76y	1.6(3)	--	1.1(1)	--	2.2(1)	1.1(1)	--	1.6(3)
Kr-87	76m	1.3(3)	--	8.9(0)	--	--	8.9(0)	--	1.3(3)
Kr-88	2.8h	4.0(3)	--	2.8(1)	--	1.0(-1)	2.8(1)	--	4.0(3)
Kr-89	3.2m	6.0(1)	--	6.7(-1)	--	--	6.7(-1)	--	6.1(1)
Xe-131m	11.8d	1.8(3)	--	1.3(1)	--	4.8(0)	1.3(1)	--	1.8(3)
Xe-133m	2.26d	4.2(3)	--	3.0(1)	--	2.1(0)	3.0(1)	--	4.3(3)
Xe-133	5.27d	3.2(5)	--	2.3(3)	2.0(0)	3.8(2)	2.3(3)	1.0(0)	3.2(5)
Xe-135m	15.6m	2.4(2)	--	2.0(0)	--	--	2.0(0)	--	2.4(2)
Xe-135	9.2h	6.6(3)	--	4.8(1)	--	5.8(-1)	4.8(1)	--	6.7(3)
Xe-137	3.9m	1.3(2)	--	1.4(0)	--	--	1.4(0)	--	1.3(2)
Xe-138	17.0m	8.3(2)	--	6.6(0)	--	--	6.6(0)	--	8.4(2)
Total		3.4(5)	--	2.5(3)	2.0(0)	4.1(2)	2.5(3)	1.0(0)	3.4(5)

[a] 4.3(4) = 4.3 x 10^4 or 43,000

[b] -- implies less than 0.1 Ci/yr

these source terms are summarized in Table 12 and are based on operating experience where possible (2, 6, 9, 51) and/or generally acceptable values (1-3, 7, 24, 26-28). Table 22 presents the source term for each of these PWR noble gas release pathways for a two unit (1000 MWe each) PWR power station. Almost 100% of the total of these sources results from the primary gases (shim bleed and shutdown degasification). Therefore, the majority of noble gas discharge control options have been designed to control these gases.

The noble gas discharge control options considered fall into five general classes in addition to the source term. The first class of options consists of pure physical holdup of primary gases (by pressurized tanks with or without the use of a recombiner) for 15, 30, 45, or 60 days and the delay of primary gases on charcoal adsorption beds, providing 15, 30, 45, and 60 days delay for xenon and 1, 2, 3, and 4 days for krypton (1, 24, 29, 32, 34, 35). Flow rate of radioactive primary gases was taken as 0.5 scfm from each unit (1.0 scfm total flow rate). Costs have been estimated from available literature (1, 13, 21, 22, 24, 36-39). Comparison of pure physical holdup with charcoal adsorption reveals virtually the same total releases for xenon holdup times of 15, 30, 45, and 60 days. For shorter holdup times, however, slightly more krypton will be released from the charcoal adsorption beds.

The second class of PWR noble gas discharge control options considered consists of treating primary gases with either a selective absorption system or cryogenic distillation. In each case, the noble gases xenon and krypton are concentrated, by means of solubility into a fluorocarbon solvent in the former case, and by temperature effects in the latter case. As these options have had limited operational experience, the cost and effectiveness estimates appear less certain. Consequently, two decontamination factors (DFs) for xenon, 1,000 and 10,000, are assumed for each of these options (1,37,41,45,46), resulting in a range of estimates for releases.

Krypton DFs are assumed to be 25% of the xenon DFs, in accordance with assumptions presented in recent safety analysis reports. However, once either set of noble gas DFs is applied to the primary system off gases, the variation in total release is small as secondary sources predominate. Cost estimates show considerable variability (1,21,22,38). This second class of noble gas discharge control options results in a release of noble gases similar to the first class of options providing 60 day delay or holdup.

A third class of discharge control options may be defined by combining the treatment systems afforded by the first two classes. However, analysis shows that release of noble gases from primary gases decreases from 51 Ci/yr to about 6.5 Ci/yr using different holdup times from 15 to 60 days, respectively; this decrease is negligible when compared to total noble gas releases of about 5,400 Ci/yr for all such system combinations. Therefore, the third class of discharge control options will be assumed to be represented by a 15 day xenon delay on charcoal adsorbers and cryogenic distillation or selective absorption which appears to be the least expensive option.

A fourth discharge control option class is defined as the virtual elimination of primary gas releases by using the cover gas recycle system.

Volatile radioisotopes are continuously removed from the primary coolant system by a constant purge of the volume control tank. This hydrogen-fission gas mixture is sent to the cover gas recycle system, where it is diluted with nitrogen, passed through a compressor and recombiner, and stored in a gas decay tank. By using many gas decay tanks, the nitrogen in the cover gas recycle system may be recycled indefinitely, allowing significant decay of all noble gases (except krypton-85) before the nitrogen in a given decay tank must be used again. As a result, primary coolant concentrations of the volatile radioisotopes are reduced. The effect, however, is more pronounced for radioisotopes with longer half-lives (1,47):

Nuclide	Half-life	Factor of Reduction in Primary Coolant Concentration
Xe-135m	15.6 m	1.0
Kr-87	76 m	1.0
Kr-85m	4.4 h	1.0
Xe-135	9.2 h	1.3
Xe-133m	2.26 d	5.0
Xe-133	5.27 d	7.0
Kr-85	10.76 y	10.0

Thus, cover gas recycle is unique in that the reduction of primary coolant concentrations results in a decrease of the releases of volatile radionuclides from all gaseous release pathways. Cost estimates are very limited (1,22).

Finally, a fifth class of discharge control options is defined as the treatment of noble gas sources other than primary gases. As shown by the PWR Source Term, Table 22, the bulk of these releases is made up of the releases from the air ejector and the auxiliary building. Because of the high flow from the auxiliary building (100,000 cfm) (2), it appears impractical at present to treat this effluent stream. However, charcoal beds may be used to delay the noble gases evolving from a PWR air ejector (35), as is commonly proposed for BWRs. Cost and effectiveness, therefore, are largely based on BWR data (1, 13, 29, 32, 35, 39). Table 23 illustrates the effectiveness of air ejector charcoal beds in reducing these noble gases, using 1, 2, 3, and 10 days xenon delay and a Xe/Kr delay ratio of 15 (1,24,29,35).

In order to place these PWR noble gas discharge control options in perspective, a summary table, Table 24, was constructed to illustrate the range of options considered, estimated costs, and activity releases. It should be noted that the use of compressor-tankage holdup is most typical of present PWR waste gas systems, although it appears more expensive than charcoal adsorption. Application of some of the more sophisticated discharge control options for primary gases results in releases overwhelmingly dominated by the secondary source contribution. Only the use of cover gas recycle for primary gases and the use of charcoal delay beds on the air ejector can reduce the contribution of these secondary release pathways of noble gases. To illustrate the change in the technology of controlling noble gas releases, PWRs formerly provided only a 15-45 day holdup for primary gases whereas presently planned PWRs typically are providing 60 day holdup. At least one PWR has proposed charcoal adsorption beds for both primary gases and air ejector effluent (35), a few have gone to the cover gas recycle system (4, 47), and at least two plants have proposed adding cryogenic charcoal adsorption systems (in which the noble gases are adsorbed on low-temperature charcoal) to existing gas decay tanks (15,57).

Table 23

Effectiveness of Charcoal Delay Beds on Air Ejector (PWR)

Nuclide	Half-Life(3	Air Ejector Source	Xe Delay: 1 Day (Kr=0.067d)	Xe Delay: 2 Days (Kr=0.133d)	Xe Delay 3 Days (Kr=0.20d)	Xe Delay 10 Days (Kr=0.67d)
		Releases (Ci/yr for 2 Units, 1000 MWe each)				
Kr-83m	1.86h	3.0(0)[a]	8.2(-1)	2.3(-1)	--[b]	--
Kr-85m	4.4h	1.6(1)	1.2(1)	9.7(1)	7.5(0)	1.3(0)
Kr-85	10.76y	1.1(1)	1.1(1)	1.1(1)	1.1(1)	1.1(1)
Kr-87	76m	8.9(0)	3.7(0)	1.5(0)	6.2(-1)	--
Kr-88	2.8h	2.8(1)	1.9(1)	1.3(1)	8.7(0)	5.2(-1)
Kr-89	3.2m	6.7(-1)	--	--	--	--
Xe-131m	11.8d	1.3(1)	1.2(1)	1.1(1)	1.0(1)	7.2(0)
Xe-133m	2.26d	3.0(1)	2.2(1)	1.6(1)	1.2(1)	1.4(0)
Xe-133	5.27d	2.3(3)	2.0(3)	1.8(3)	1.6(3)	6.1(2)
Xe-135m	15.6m	2.0(0)	--	--	--	--
Xe-135	9.2h	4.8(1)	7.9(0)	1.3(0)	2.1(-1)	--
Xe-137	3.9m	1.4(0)	--	--	--	--
Xe-138	17.0m	6.6(0)	--	--	--	--
		2.5(3)	2.1(3)	1.9(3)	1.6(3)	6.4(2)

Effectiveness based on 10 cfm air in-leakage, Xe/Kr delay ratio of 15, and following amounts of charcoal:

Xe Delay	Charcoal (tons)
1d	3.6
2d	7.2
3d	10.8

[a] 3.0(0) = 3.0 x 10^0 or 3.0
[b] -- implies less than 0.1 Ci/yr

Table 24

SUMMARY TABLE: PWR NOBLE GAS DISCHARGE CONTROL OPTIONS
(2 UNITS, 1000 MWE EACH)

CLASS	PWR NOBLE GAS DISCHARGE CONTROL OPTION	DAYS HOLDUP OR PROCESS DF	ESTIMATED CAPITAL COST	ESTIMATED ANNUAL COST	RELEASE (Ci/yr)			
					Kr-85	Kr-88	Xe-133	Total[a]
0	No Treatment	None	$ 0	$ 0	1.6(3)	4.0(3)	3.2(5)	3.4(5)
1A-15	Charcoal Adsorption	15d Xe;1d Kr	$360,000	$ 60,000	"	6.7(1)	4.9(4)	5.2(4)
1A-30	Charcoal Adsorption	30d Xe;2d Kr	$540,000	$ 90,000	"	5.6(1)	1.1(4)	1.3(4)
1A-45	Charcoal Adsorption	45d Xe;3d Kr	$720,000	$120,000	"	"	5.8(3)	7.9(3)
1A-60	Charcoal Adsorption	60d Xe;4d Kr	$900,000	$150,000	"	"	5.1(3)	7.1(3)
1B-15	Compressed Tank Holdup (w/wo recombiner)	15d	$850,000/ /$500,000	$270,00/ /$164,000	"	"	4.9(4)	5.2(4)
1B-30	" "	30d	$900,000/ /$600,000	$280,000/ /$190,000	"	"	1.1(4)	1.3(4)
1B-45	" "	45d	$950,000/ /$700,000	$290,000/ /$225,000	"	"	5.8(3)	7.9(3)
1B-60	" "	60d	$1,000,000/ /$800,000	$300,000/ /$250,000	"	"	5.1(3)	7.1(3)
2A	Cryogenic Distillation or Selective Adsorption	Xe DF = 1,000	$1,500,000	$600,000	5.0(1)	6.4(1)	5.3(3)	5.7(3)
2B	" "	Xe DF = 10,000	$1,500,000	$600,000	4.5(1)	5.7(1)	5.0(3)	5.4(3)
3	1A-15 + 2A	15d Xe + Xe DF = 1,000	$1,860,000	$660,000	5.0(1)	5.6(1)	5.0(3)	5.4(3)
4	Cover Gas Recycle	b	$2,000,000	$580,000	4.4(0)	5.6(1)	7.1(2)	9.4(2)
5A	Air Ejector Charcoal Adsorption + Class 1A	2d Xe[c]	$1,260,000	$210,000	1.6(3)	5.2(1)	4.9(4)	5.2(4)
5B	" + Class 2A	2d Xe[c]	$2,400,000	$750,000	5.0(1)	4. (1)	4.8(3)	5.1(3)
5C	" + Class 3	2d Xe[c]	$2,760,000	$810,000	5.0(1)	4.1(1)	4.5(3)	4.8(3)
5D	" + Class 4	2d Xe[c]	$2,900,000	$730,000	4.4(0)	4.1(1)	6.5(2)	7.9(2)

[a] From all noble gas radionuclides

[b] Virtually complete holdup of primary gases may be achieved; releases from secondary pathways are also reduced by about a factor of 5.8.

[c] 2 days xenon delay for air ejector noble gases plus appropriate primary gas holdup or DF indicated by class of treatment.

Boiling Water Reactors (BWRs)

As in the PWR, radioactive isotopes of krypton and xenon, the noble gases of greatest concern, are generated inside the fuel rods. These gaseous radionuclides escape through the fuel rod cladding defects and enter the reactor coolant system. There are many leakage pathways which may allow the radioactive isotopes of krypton and xenon to reach the environment. These release pathways may be broken down as follows:

(1) Primary Gases
    Condenser Air Ejector
    Turbine Gland Seal
    Mechanical Vacuum Pump (at startup)

(2) Building Ventilation
    Reactor Building
    Radwaste Building
    Turbine Building

Whereas the release through the air ejector is intimately related to plant operation, other pathways are more or less unplanned since they occur as leakages from plant components.

In order to determine the effectiveness of various BWR noble gas discharge control options, the source term for each release pathway was determined. The releases from the condenser air ejector and turbine gland seal are based on estimates made by the nuclear industry (29,30). It should be noted that the air ejector source term incorporates a nominal 30-minute delay. This is typical of all previous BWR plants (1,29). Primary coolant concentrations corresponding to the 30-minute air ejector source term and the assumptions specified for leakage conditions in various buildings, Table 11, for the basis for the noble gas releases for

Table 25

BWR Source Term: 2 Units, 1,000 MWe Each (Ci/yr)

Radio-Nuclide	Half-life (31)	Air Ejector Effluent (29,30)	Turbine Gland Seal Effluent (30)	Reactor Building	Radwaste Building	Turbine Building	Mechanical Vacuum Pump	Totals
Kr-83m	1.86h	1.5(5)[a]	3.5(2)	---[b]	---	2.9(1)	---	1.5(5)
Kr-85m	4.4h	2.8(5)	6.1(2)	---	---	4.9(1)	---	2.8(5)
Kr-85	10.76y	7.6(2)	2.4(0)	---	---	4.2(-1)	---	7.6(2)
Kr-87	76m	7.6(5)	1.9(3)	---	---	1.7(2)	---	7.6(5)
Kr-88	2.80h	9.1(5)	2.0(3)	---	---	1.7(2)	---	9.1(5)
Kr-89	3.2m	9.1(3)	3.8(3)	---	---	1.0(3)	---	1.4(4)
Kr-90	33.0s	---	3.1(3)	---	---	2.0(2)	---	3.3(3)
Xe-131m	11.8d	7.6(2)	2.0(0)	---	---	1.2(0)	---	7.6(2)
Xe-133m	2.26d	1.5(4)	3.0(1)	---	---	2.3(0)	---	1.5(4)
Xe-133	5.27d	4.1(5)	8.6(2)	---	---	6.6(1)	4.4(3)	4.1(5)
Xe-135m	15.6m	3.5(5)	2.4(3)	---	---	2.1(2)	---	3.5(5)
Xe-135	9.2h	1.1(6)	2.2(3)	---	---	1.8(2)	6.6(2)	1.1(6)
Xe-137	3.9m	3.4(4)	1.1(4)	---	---	1.3(3)	---	4.6(4)
Xe-138	17.0m	1.1(6)	8.6(3)	---	---	5.9(2)	---	1.1(6)
Xe-139	43.0s	---	4.7(3)	---	---	9.9(2)	---	5.7(3)
Totals		5.1(6)	4.2(4)			5.0(3)	5.1(3)	5.1(6)

[a] 1.5(5) = 1.5 x 10^5 or 150,000

[b] --- = < 0.1 Ci/Yr

building ventilation sources. Table 25 presents the source term for each of the release pathways discussed above for a two-unit, 1000 MWe each, BWR power station. Almost 100% of the total release is from the condenser air ejector. As a result, all but one of the noble gas discharge control options considered are designed to reduce this release pathway.

BWR noble gas discharge control options may be divided into five classes; four of these reduce air ejector releases while one option has been considered for eliminating turbine gland seal effluents. The first class consists of a single option, physical holdup of air ejector offgases (via recombiner-compressor-tankage) for one day; this option has been proposed for at least two BWRs (19,40). These plants have stacks, however, unlike more recent designs. A condenser air inleakage of 10 scfm is assumed; smaller air inleakage can increase holdup time proportionately while greater leakage would decrease holdup. Costs are based on available data (1,21,23,38-41).

The second class of discharge control options consists of the use of recombiners and varying amounts of charcoal (at 77° F) to achieve xenon holdup times of 10, 20, 40, and 60 days. However, krypton delay times of only 1, 2, 3, and 4 days, respectively, are achieved, based on measured and suggested values of the Xe/Kr delay ratio (1,24,29,35). Condenser air inleakage was assumed to be 10 scfm, based on measured and suggested values (1,29,32-34). Cost estimates appear more certain for this class of options than any of the others and are based primarily upon nuclear industry estimates (1,13,24,39,42,43). However, this second class of control options achieves a lower release rate at costs comparable to the first class, although releases are made from a plant vent rather than a 100 m stack.

Alternatively, a third class of discharge control options may be formed by considering the processing of the air ejector effluent through a selective absorption system or a cryogenic distillation system. In each case, the noble gases xenon and krypton are concentrated, by means of solubility into a fluorocarbon solvent in the former case, and by temperature effects in the latter case. As these options have had limited operational experience, cost and effectiveness estimates show a wide variation. Consequently, two decontamination factors (DFs) for xenon, 1,000 and 10,000, are assumed for each of these options (1,37,41,45,46), resulting in a range of estimates for releases. Krypton DFs are assumed at 25% of xenon DFs. However, since secondary release pathways dominate total releases with either DF, the variation in total releases is small. Cost estimates exhibit a wide variability (1,13,41). Either of these options, selective absorption or cryogenic distillation, provides about the same activity reduction as a 20-40 day delay of xenon on charcoal beds.

By combining the treatment options in the first three classes above, a fourth set of discharge control options for BWR noble gas air ejector releases may be defined. Since secondary source terms now dominate, the variation in total release by using different combinations of classes 1, 2, and 3 is minimal. Therefore, the combination of a ten-day xenon delay on charcoal (0.67 day delay for krypton) and either a selective absorption system or cryogenic distillation system will be taken to represent this fourth class of noble gas discharge control options.

The fifth class of BWR noble gas discharge control options is the only option chosen to reduce noble gas emissions from a source other than the condenser air ejector, namely, the turbine gland seal exhaust.

Examination of the BWR source term indicates that other than the air ejector only the turbine gland seal, turbine building ventilation, and operation of the mechanical vacuum pump contribute significantly to noble gas releases. However, a source of nonradioactive steam may be used to block the release of radioactive gases from the turbine gland seal (48), eliminating this source of noble gas release. Cost estimates are based on very limited data (1,24,25).

In order to place these BWR noble gas discharge control options in perspective, a summary table, Table 26, has been constructed to illustrate the range of options considered, estimated costs, and activity releases. In the case of the clean steam options, the only alternative considered for treating a secondary source of noble gases, only those options are shown that illustrate a large change in activity releases when used in conjunction with a given primary gas treatment option. Application of some of the more sophisticated discharge control options for primary gases results in releases overwhelmingly made up of the secondary source contribution. Although presently operating BWRs employ essentially a Class 0 treatment, almost all are planning to retrofit equipment to achieve Class 2-10 (charcoal adsorption) releases within 1-3 years. A few BWRs presently planned have proposed cryogenic distillation systems to treat the primary off-gas from the air ejector as well as clean steam systems to eliminate the noble gases from the turbine gland seal (58,59).

Table 26

SUMMARY TABLE: BWR NOBLE GAS DISCHARGE CONTROL OPTIONS
(2 UNITS, 1000 MWE EACH)

CLASS	DISCHARGE CONTROL OPTION	DAYS HOLDUP OR PROCESS DF	ESTIMATED CAPITAL COST	ESTIMATED ANNUAL COST	RELEASE (Ci/yr)				
					Kr-85	Kr-88	Xe-133	Xe-138	Total[a]
0	Source Term[b]	-----	$3,200,000	$ 600,000	7.6(2)	9.1(5)	4.1(5)	1.1(6)	5.1(6)
1	Recombiner-Holdup-Stack	1d	$6,500,000	$1,400,000	"	4.6(3)	3.6(5)	9.2(3)	6.0(5)
2-10	Recombiner Charcoal Adsorption	10d Xe	$5,600,000	$1,200,000	"	1.9(4)	1.1(5)	"	2.0(5)
2-20	" "	20d Xe	$6,000,000	$1,330,000	"	2.5(3)	3.5(4)	"	8.5(4)
2-40	" "	40d Xe	$7,000,000	$1,620,000	"	2.2(3)	7.5(3)	"	5.4(4)
2-60	" "	60d Xe	$8,000,000	$1,910,000	"	"	5.5(3)	"	5.2(4)
3A	Cryogenic Distillation or Selective Absorption	Xe DF = 1,000	$7,000,000	$1,400,000	7.8(0)	5.8(3)	5.7(3)	1.0(4)	6.3(4)
3B	" "	Xe DF =10,000	$7,000,000	$1,400,000	3.1(0)	2.5(3)	5.4(2)	9.3(3)	5.3(4)
4	2-10 + 3A	10d Xe+ Xe DF= 1,000	$8,8000,000	$2,600,000	5.8(0)	2.2(3)	5.4(3)	9.2(3)	5.2(4)
5A	CLEAN STEAM + CLASS 1	c	$7,500,000	$1,600,000	7.6(2)	2.6(3)	3.6(5)	5.9(2)	5.6(5)
5B	CLEAN STEAM + CLASS 2-10	c	$6,600,000	$1,400,000	7.6(2)	1.7(4)	1.1(5)	5.9(2)	1.6(5)
5C	CLEAN STEAM + CLASS 2-60	c	$9,000,000	$2,110,000	7.6(2)	1.7(2)	4.6(3)	5.9(2)	1.1(4)
5D	CLEAN STEAM + CLASS 3A	c	$8,000,000	$1,600,000	3.4(0)	3.7(3)	4.9(3)	1.7(3)	2.1(4)
5E	CLEAN STEAM + CLASS 4	c	$9,800,000	$2,800,000	3.4(0)	1.8(2)	4.6(3)	5.9(2)	1.0(4)

[a] Total is sum of all noble gas radioisotope activities released

[b] Illustrates effects of two 1,000 MWe BWRs operating with presently operating air ejector off-gas systems (30 minute delay pipe and 100 m stack)

[c] Clean steam may virtually eliminate noble gases and radioiodines from the turbine gland seal; appropriate primary gas holdup, delay, or DF for each class must be considered for each combination.

RADIOIODINE DISCHARGE CONTROL OPTIONS

Pressurized Water Reactors (PWRs)

Radioactive isotopes of iodine are generated inside the fuel rods of pressurized water reactors. These volatile isotopes escape through the fuel rod cladding defects and enter the primary coolant system. Due to the leakages of primary coolant and/or various plant operations, radioiodines may be released to the environment. These release pathways may be broken down as follows:

(1) Primary Gases
  - Shutdown Degasification
  - Shim Bleed

(2) Secondary System Gases
  - Air Ejector Exhaust
  - Gland Seal Exhaust
  - Steam Generator Blowdown Tank Vent

(3) Building Ventilation
  - Containment Purge
  - Auxiliary Building
  - Turbine Building

In order to determine the effectiveness of various PWR radioiodine control options, the source term for each of the release pathways detailed above was calculated and is shown in Table 27. However, radioiodine evolving from the primary gases is not included since the noble gas treatment options currently planned for PWR primary gases (60-day holdup, cryogenic distillation, selective absorption, or cover gas recycle) should effectively minimize this pathway of radioiodine release. Assumptions for calculating the sources of radioiodine release are shown in Table 12. Because of the uncertainty in the chemical form of radioiodine released, two cases are chosen for consideration, namely, that all radioiodine released is either in elemental or organic form.

Aside from the uncertainty in chemical form, uncertainty also exists relative to the decontamination factor (DF) achieved in practice by charcoal adsorbers. Existing test data would indicate a DF of charcoal adsorbers for elemental radioiodine on the order of 100 - 10,000 and for methyl (organic) iodide, a DF of 4-1000 (50). As these tests were generally performed under controlled laboratory conditions, these DFs may not be representative of the conditions to be experienced by charcoal adsorbers in the various types of reactor gaseous effluent streams. Many factors (such as chemical form of iodine, relative humidity, atmospheric contaminants, leak-tightness of adsorber assembly, etc.) may combine to degrade the DFs reported above. A comprehensive investigation of the effectiveness of charcoal for removing radioiodine has been recommended (49). As a result, DFs for charcoal adsorbers used on reactor plant gaseous effluents has been taken to be 10. A DF of 100 is used for deep bed charcoal adsorbers. The use of an internal recirculation charcoal adsorber (commonly referred to as a "kidney") in a PWR containment, through which a fractional volume of the building air is passed per unit time, decreases the concentration of iodine-131 and iodine-133 in the containment atmosphere by factors of 3 and 7, respectively. This is based on an 8,000 cfm flow rate for 16 hours of cleanup of 70% of the containment atmosphere after a 90-day buildup. Routing of the steam generator blowdown tank vent to the main condenser effectively minimizes this pathway of release at a PWR, partly because of the high partition factor obtained in the condenser. The use of clean steam on the turbine gland seal effectively eliminates this pathway of release and has been proposed for BWRs only. In summary, the following discharge control options were considered for a PWR:

(1) Steam Generator Blowdown Vent to Main Condenser
(2) Charcoal Kidney Adsorber inside Containment
(3) Steam Jet Air Ejector Charcoal Adsorber
(4) Auxiliary Building Charcoal Adsorber
(5) Auxiliary Building Deep Bed Charcoal Adsorber
(6) Clean Steam: Valves > 2.5" diameter in Turbine Building
(7) Clean Steam on Turbine Gland Seal

Tables 28 and 29 detail these treatment options, generally added in order of increasing cost per curie of iodine-131 eliminated, estimated

Table 27

PWR Radioiodine Source Term: 2 Units, 1,000 MWe Each (Ci/yr)

	ELEMENTAL		ORGANIC	
	I-131	I-133	I-131	I-133
STEAM GENERATOR BLOWDOWN TANK VENT	0.56	0.34	11.0	6.6
STEAM JET AIR EJECTOR	0.10	0.062	10.0	6.2
GLAND SEAL EFFLUENT	0.0002	0.0001	0.0002	0.0001
CONTAINMENT	0.70	0.068	7.0	0.68
AUXILIARY BUILDING	0.11	0.13	13.0	16.0
TURBINE BUILDING	0.04	0.024	0.04	0.024
TOTALS	1.51	0.624	41.0	29.5
PRIMARY GASES[a]	0.0031	3.0×10^{-5}	0.31	0.003

[a] Source term given only for comparative purposes; present and future treatment systems will reduce this source to negligible levels (i.e., less than 0.001 Ci/yr).

Table 28

Annual Costs for Radioiodine (Elemental) Removal From PWR Gaseous Effluents
(2 Units, 1000 MWe each)

Control Option Added	System Designation	Estimated Capital Cost (Cumulative)	Estimated Annual Cost (Cumulative)	Radioiodine Release (Ci/yr)		
				I-131	I-133	Total
None [a]	PGIE-1	$ 0	$ 0	1.510	0.624	2.134
Containment Kidney	PGIE-2	$ 700,000	$ 120,000	1.044	0.566	1.610
Steam Generator Blowdown vented to Condenser	PGIE-3	$ 950,000	$ 160,000	0.484	0.226	0.710
Auxiliary Building Charcoal Adsorber [b]	PGIE-4	$2,950,000	$ 460,000	0.175	0.100	0.275
Air Ejector Charcoal Adsorber	PGIE-5	$3,350,000	$ 560,000	0.085	0.044	0.129
Upgrade to Deep Bed Charcoal Adsorber: Auxiliary Building [b]	PGIE-6	$4,130,000	$ 860,000	0.054	0.032	0.086
Clean Steam: Valves > 2.5" Diameter	PGIE-7	$5,930,000	$1,160,000	0.022	0.013	0.035
Gland Seal Clean Steam	PGIE-8	$6,530,000	$1,360,000	0.021	0.012	0.033

[a] Does not include radioiodine from primary system gases (shutdown degasification, shim bleed) as these are effectively removed by gaseous waste treatment systems.

[b] Containment purge is also routed through this adsorber.

Table 29

Annual Costs for Radioiodine (Organic) Removal From PWR Gaseous Effluents
(2 Units, 1000 MWe each)

Control Option Added	System Designation	Estimated Capital Cost (Cumulative)	Estimated Annual Cost (Cumulative)	Radioiodine Release (Ci/yr)		
				I-131	I-133	Total
None[a]	PGIO-1	$ 0	$ 0	41.040	29.504	70.544
Containment Kidney	PGIO-2	$ 700,000	$ 120,000	36.374	28.921	65.295
Steam Generator Blowdown vented to Condenser	PGIO-3	$ 950,000	$ 160,000	25.374	22.321	47.695
Air Ejector Charcoal Adsorber	PGIO-4	$1,530,000	$ 260,000	16.374	16.741	33.115
Auxiliary Building Charcoal Adsorber[b]	PGIO-5	$3,530,000	$ 560,000	2.574	2.254	4.828
Upgrade to Deep Bed Charcoal Adsorber: Auxiliary Building[b]	PGIO-6	$4,130,000	$ 860,000	1.194	0.805	1.999
Clean Steam: Valves >2.5" Diameter	PGIO-7	$5,930,000	$1,160,000	1.162	0.786	1.948
Gland Seal Clean Steam	PGIO-8	$6,530,000	$1,360,000	1.161	0.786	1.947

[a] Does not include radioiodine from primary system gases (shutdown degasification, shim bleed) as these are effectively removed by gaseous waste treatment systems.

[b] Containment purge is also routed through this adsorber.

costs (1, 16, 24, 25), and estimated releases from a two-unit (1000 MWe each) PWR power station. In any case, the uncertainty associated with the costs of progressively improved treatment increases tremendously beyond the first three or four equipment additions.

Whereas many PWRs formerly included only a charcoal adsorber on the primary gas decay tank dishcarge and a containment kidney adsorber, present design typically includes these as well as venting the steam generator blowdown tank to the main condenser and auxiliary building charcoal adsorbers. At least one PWR has planned to treat the air ejector effluent via charcoal delay beds (35) but this is as much meant to reduce noble gas releases as radioiodine releases. Finally, an overall reduction by perhaps a factor of 2 can be obtained by those PWRs using the cover gas recycle system for the control of primary gas radioactivity release (1, 47).

Boiling Water Reactors (BWRs)

Radioactive isotopes of iodine are also generated inside the fuel rods of boiling water reactors and may escape to the reactor coolant system through defects in the fuel rod cladding. Due to leakages of reactor coolant, and/or various plant operations, radioiodines may be released to the environment. These release pathways may be broken down as follows:

(1) Primary Gases

Condenser Air Ejector

Turbine Gland Seal

(2) Building Ventilation

Turbine Building

Reactor Building

Radwaste Building

The source term for each of the release pathways detailed above was calculated and is shown in Table 30. However, the noble gas treatment options currently planned for the BWR air ejector source term should effectively minimize this release pathway. Assumptions for calculating the sources of radioiodine release are shown in Table 11. Because of the uncertainty in the chemical form of radioiodine released, two cases are chosen for consideration, namely, that all radioiodine released is either in elemental or organic form.

Six radioiodine discharge control options were considered for BWRs:

(1) Clean Steam: Valves > 2.5" diameter (Turbine Building)

(2) Turbine Building Charcoal Adsorber

(3) Turbine Building Deep Bed Charcoal Adsorber

(4) Reactor Building Charcoal Adsorber

(5) Radwaste Building Charcoal Adsorber

(6) Turbine Gland Seal Clean Steam

Due to the uncertainty associated with charcoal adsorber DFs, as previously discussed, a value of 10 has been used. For a deep bed charcoal adsorber, an incremental DF of 10 is used. The use of clean steam on the gland seal effectively eliminates this pathway of release. A DF of 5 is more or less presumed for the use of clean steam on valves greater than 2.5 inches in diameter in the turbine building. It should be noted that no credit is taken for the use of the standby gas treatment system in decontaminating the reactor building releases.

Tables 31 and 32 detail these treatment options, considered as individual successive equipment additions (in order of increasing cost per curie of iodine-131 eliminated), estimated costs (1, 24, 25), and estimated releases from a two-unit (1000 MWe each) BWR power station. Uncertainty associated with the cost of systems increases rapidly with the addition of equipment.

Whereas BWRs typically included no treatment for radioiodine releases, BWRs of current design are incorporating combinations of such features as supplying clean steam to valves in the turbine building, a deep bed turbine building charcoal adsorber, and a turbine gland seal clean steam system.

Table 30

BWR Radioiodine Source Term: 2 Units, 1,000 MWe Each (Ci/yr)

	ELEMENTAL		ORGANIC	
	I-131	I-133	I-131	I-133
GLAND SEAL	0.0058	0.0330	0.0058	0.0330
REACTOR BUILDING	0.0170	0.0990	17.2000	99.0000
RADWASTE BUILDING	0.0014	0.0040	0.1200	0.6800
TURBINE BUILDING	1.0000	5.7000	1.0000	5.7000
TOTALS	1.024	5.836	28.326	105.413
PRIMARY GASES[a]	29.0	170.	~150.	~850.

[a] Source term given only for comparative purposes; present and future treatment systems will reduce this source to negligible levels (i.e., less than 0.001 Ci/yr)

Table 31

Annual Costs for Radioiodine (Elemental) Removal From BWR Gaseous Effluents
(2 units, 1000 MWe each)

Control Option Added	System Designation	Estimated Capital Cost (Cumulative)	Estimated Annual Cost (Cumulative)	Radioiodine Release (Ci/yr)		
				I-131	I-133	Total
Presently Operating[a]	BGIE-1A	$ 0	$ 0	(30.0)	(176.)	(206.)
None[b]	BGIE-1	$ 0	$ 0	1.02	5.84	6.86
Clean Steam: Valves > 2.5" Diameter	BGIE-2	$1,800,000	$ 300,000	0.224	1.28	1.50
Turbine Building Charcoal Adsorber	BGIE-3	$4,300,000	$ 750,000	0.044	0.250	0.295
Upgrade to Deep Bed Charcoal Adsorber: Turbine Building	BGIE-4	$5,100,000	$1,200,000	0.026	0.147	0.173
Reactor Building Charcoal Adsorber	BGIE-5	$7,100,000	$1,600,000	0.017	0.058	0.075
Radwaste Building Charcoal Adsorber	BGIE-6	$7,350,000	$1,640,000	0.010	0.055	0.065
Turbine Gland Seal Clean Steam	BGIE-7	$7,950,000	$1,840,000	0.004	0.022	0.026

[a] Illustrates projected effects of two 1,000 MWe BWRs operating with presently used off-gas system (i.e., 30 minute delay and 100 m stack for air ejector noble gases).

[b] Reflects source term for sources other than air ejector as "augmented" BWR noble gas treatment systems (charcoal adsorption, selective absorption, or cryogenic distillation) will effectively remove air ejector radioiodines.

Table 32

Annual Costs for Radioiodine (Organic) Removal From BWR Gaseous Effluents
(2 Units, 1000 MWe each)

Control Option Added	System Designation	Estimated Capital Cost (Cumulative)	Estimated Annual Cost (Cumulative)	Radioiodine Release (Ci/yr)		
				I-131	I-133	Total
Presently Operating[a]	BGIO-1A	$ 0	$ 0	(178.)	(955.)	(1133.)
None[b]	BGIO-1	$ 0	$ 0	28.3	105.	135
Reactor Building Charcoal Adsorber	BGIO-2	$2,000,000	$ 400,000	2.85	16.3	19.2
Upgrade to Deep Bed Charcoal Adsorber: Reactor Building	BGIO-3	$2,500,000	800,000	1.30	7.40	8.70
Radwaste Building Charcoal Adsorber	BGIO-4	$2,750,000	840,000	1.19	6.79	7.98
Clean Steam: Valves >2.5" Diameter	BGIO-5	$4,550,000	$1,140,000	0.390	2.23	2.62
Turbine Building Charcoal Adsorber	BGIO-6	$7,050,000	$1,590,000	0.210	1.21	1.42
Turbine Gland Seal Clean Steam	BGIO-7	$7,650,00	$1,790,000	0.204	1.17	1.37

[a] Illustrates projected effects of two 1,000 MWe BWRs operating with presently used off-gas system (i.e., 30 minute delay and 100 m stack for air ejector noble gases).

[b] Reflects source term for sources other than air ejector as "augmented" BWR noble gas treatment systems (charcoal adsorption, selective absorption, or cryogenic distillation) will effectively remove air ejector radioiodines.

DETERMINATION OF POPULATION RADIATION EXPOSURE

The estimation of potential health risks associated with radioactivity releases from nuclear power reactors requires an assessment of the radiation exposure resulting from these releases. This dose assessment is a difficult and complex task. The complexity of the dose assessment results from: (a) the number of different radionuclides produced and released from the nuclear reactor (there are at least 100 major radionuclides and over 300 radionuclides of lesser significance); (b) the multiplicity of release paths from the facility; (c) the number of environmental vectors which can convey the radionuclides to man; and (d) the number of body organs which may be irradiated by a given radionuclide.

Detailed studies of radionuclide effluents, exposure pathways, and radiation doses have indicated that this complex situation can be simplified by consideration of the most critical pathways and principal radionuclides which contribute significantly to the radiation dose. Both calculational studies (53) and environmental measurements (9,51) have indicated that the principal radionuclides which contribute to radiation exposure from nuclear reactor effluents can be reduced to approximately two dozen in number which interact via the exposure pathways shown in Table 33.

The radioiodines (principally I-131 and I-133) are of importance because of the relatively large yield in uranium fission and the high affinity of the thyroid gland for iodine. The major exposure pathways for radioiodine are air inhalation, ingestion of drinking water, fresh milk, beef, and lamb (53).

Cesium isotopes (Cs-134 and Cs-137) are also produced in significant quantities and contribute to the radiation dose received by the total body, bone, liver, and gastrointestinal (GI) tract. The principal exposure pathways involved are drinking water and consumption of fish and shellfish.

The noble gases krypton and xenon have many radioactive isotopes which are formed in fission (see Table 5). These inert gases are important because of their fission yields and half-lives. The only important source of exposure is external whole body irradiation by the gamma-emitting radionuclides in a cloud and the submersion skin dose from beta emitters.

Table 33

Principal Exposure Pathways for Radiation Exposure from Nuclear Reactor Effluents

Radionuclide	Discharge Mode	Principal Exposure Pathways	Critical Organ
Radioiodine	Airborne	Ground deposition - external irradiation	Whole body
		Air inhalation	Thyroid gland
		Grass-cow-milk	" "
		Leafy vegetables	" "
	Water	Drinking water	" "
		Fish consumption	" "
		Shellfish	" "
Tritium	Airborne	Air inhalation and transpiration	Whole body
		Submersion	Skin
	Water	Drinking water	Whole body
		Food consumption	" "
Noble Gases	Airborne	External irradiation	Whole body and Skin
Cesium	Airborne	Ground deposition - external irradiation	Whole body
		Grass-cow-milk	" "
		Grass-meat	" "
		Inhalation	" "
	Water	Sediments - external irradiation	" "
		Drinking water	" "
		Fish consumption	" "
Transition metals (Fe, Co,Ni,Zn,Mn)	Water	Drinking water	G.I. Tract
		Shellfish consumption	" "
		Fish consumption	" "
Direct Radiation	-	External irradiation	Whole body

Estimation of Radiation Doses from Liquid Effluent Releases

There are two principal pathways for radionuclides released as liquid effluents to reach man: ingestion of drinking water and ingestion of aquatic or marine foods (principally fish and shellfish). Other exposure pathways such as submersion (swimming), use of water for irrigation, boating, etc. are generally less significant and were not considered.

The radiation dose [equivalent] rate delivered by a given radionuclide which is ingested via water or food consumption can be calculated from the following expression:

$$\dot{D}E = GP\,(DICF)Q \quad ,$$

where $\dot{D}E$ is the dose [equivalent] rate in mrem/year,

P is a pathway transfer factor relating human intake to the radionuclide concentraton in water (pCi/year per pCi/liter),

DICF is the dose [equivalent] rate delivered per unit intake (mrem/year per pCi/year), and

Q is the annual release rate in curies per year.

The constant G is related to the dilution afforded by the condenser cooling water flow, V, and the conversion factor from curies to picocuries:

$$G = \frac{10^{12}\ \text{pCi/Ci}}{V\ \text{liters/year}}$$

The cooling water volume, V, is calculated for a 1 GWe plant operating for 0.8 years at a flow rate of 800,000 gallons per minute for once-through cooling:

$$V = \frac{0.8\left(\frac{yr}{yr}\right)\left(8 \times 10^5 \ \frac{\text{Gallons}}{\text{Minute}}\right)\left(5.256 \times 10^5 \ \frac{\text{Minutes}}{\text{Year}}\right)}{.2642 \ (\text{gallons/liter})} = 1.27 \times 10^{12} \ \frac{\text{liters}}{\text{year}}$$

and at 22,000 gallons per minute for the blowdown from an augmented cooling system (cooling towers):

$$V = \frac{0.8\left(\frac{yr}{yr}\right)\left(2.2 \times 10^4 \ \frac{\text{Gallons}}{\text{Minute}}\right)\left(5.256 \times 10^5 \ \frac{\text{Minutes}}{\text{Year}}\right)}{.2642 \ (\text{Gallons/liter})} = 3.47 \times 10^{10}$$

For once-through-cooling G has the value of 0.785 and it is 28.8 for plants with cooling towers.

The pathway transfer factor, P, for the water ingestion pathway is defined by:

$$P \left(\frac{\text{pCi/year}}{\text{pCi/liter}}\right) = I_w F(DF) \exp\left[-\lambda \tau_w\right] \quad ,$$

where I_w is the annual drinking water consumption rate in liters per year, F is a factor to correct for removal of radionuclides by conventional purification process at water intakes, DF is a factor to account for dilution between the effluent discharge canal and the water intake, λ is the radiological decay constant of the radionuclide, and τ_w is the time interval between discharge and consumption of water. The removal factor, F, is given for various radionuclides in Table 34. The dilution factor permits a factor of two reduction in concentration prior to consumption of water by an individual, a factor of one hundred reduction prior to reaching the water supply for large population groups for the lake site, and a factor of twenty reduction for the river site.

The pathway transfer factor, P, for ingestion of fish and shellfish has the following derivation:

$$P \left(\frac{\text{pCi/year}}{\text{pCi/liter}}\right) = I_F R(DF)(CF) \exp[-\lambda\tau_f] \quad ,$$

where:

I_F is the food ingestion rate in kilograms/year,

R is a preparation loss factor to account for removal of radionuclides during food cleaning and cooking,

DF is the dilution factor between the effluent discharge point and the fish/shellfish,

CF is a concentration factor which relates uptake by the organism to the concentration in water (pCi/kilogram per pCi/liter = liter/kilogram), and

τ_f is the time between effluent release and consumption of fish. The concentration factors for fish and shellfish (crabs, lobsters, clams, oysters, etc.) vary for different radionuclides. They are also somewhat dependent on the concentrations of chemically similar stable elements in the water. Representative values are presented for marine (seacoast site) and freshwater (river and lakesites) species in Table 34. The values for the dilution factors and intake rates are presented in Table 35 along with the values for the other parameters.

The dose [equivalent]-intake conversion factor, DICF, is given by:

$$\text{DICF}\left(\frac{\text{mrem/year}}{\text{pCi/year}}\right) = \frac{kf[\bar{E}(QF)]\ T_E}{m} \left[1 - \exp\left(-\frac{\ln 2t}{T_E}\right)\right] \quad ,$$

where:

f	is the fraction of the intake which reaches the critical organ from inhalation (f_a) or ingestion (f_w),
$[\bar{E}(QF)]$	is the product of the effective energy per disintegration (MeV/disintegration) and the quality factor of the emitted radiation (QF) [The quality factor is the conversion between the dose equivalent (in rem) and the absorbed dose (in rad) and, consequently has units of rem/rad. For beta (β) and gamma (γ) radiation, QF = 1.0 and for alpha radiation, QF = 10.],
m	is the mass of the critical organ (grams),
t	is the duration of the exposure (days), and
T_E	is the effective elimination half-time (days) for the biological elimination from the critical organ and loss by radioactive decay.

The constant k has the value 0.074 (gram-rad-disintegrations per MeV-pCi-day). It is obtained as follows:

$k = 1.443$ (3.7×10^{-2} disintegrations/pCi-second) x (8.64×10^{4} seconds/day) x (10^{3} mrem per rem) x (1.602×10^{-6} erg/MeV) ÷ (100 ergs/Gram-rad);

the 1.443 factor is the inverse of the natural logarithm of 2. Values for m, T_E, $f[f_a$ or $f_w]$ and $[\bar{E}(QF)]$ were taken from reference (65) with two exceptions: the iodine values (Table 38) were computed for the individuals in four age groups based on the parameters shown in Table 37 and the DICF for tritium was computed using a quality factor of 1.0 instead of the 1.7 value used in (65).

Table 34

Radionuclide Dependent Factors for Liquid Effluent Dose Calculations

Radionuclide	Fraction Remaining After Treatment at Water Intake (F)(54)	Concentration Factors (CF in pCi/kg per pCi/liter) Marine(79,80) Fish	Marine(79,80) Shellfish	Freshwater(80) Fish	Freshwater(80) Shellfish
Tritium (H-3)	1.0	1.0	1.0	1.0	1.0
Mn-54	0.2	3,000	5,000	25	40,000
Fe-55	.2	1,000	20,000	300	3,200
Fe-59	.2	1,000	20,000	300	3,200
CO-58	.2	100	1,000	500	1,500
Co-60	.2	100	1,000	500	1,500
Sr-89	.2	1	6	40	700
Sr-90	.2	1	6	40	700
Y-91	.2	30	1,000	100	1,000
Zr-95	.3	30	1,000	100	1,000
Nb-95	.3	100	100	30,000	100
Mo-99	.8	10	10	100	100
Ru-103	.2	3	2,000	100	2,000
Ru-106	.2	100	2,000	5,000	2,000
I-131	.8	500	50	1	25
I-133	.8	200	50	1	25
Cs-134	.8	30	20	1,000	1,000
Cs-137	.8	30	20	1,000	1,000
Ce-141	.2	30	1,000	100	1,000
Ce-144	.2	30	1,000	100	1,000
Pr-144	.2	30	1,000	100	1,000

Table 35

Parameters Used for the Calculation of the Radiation Dose from Liquid Effluents

Site Parameters	Seacoast	River	Lake	Reference
Dilution Factor to Receptor (DF)				
Critical Individual	0.5	0.5	0.5	
Population Average	.01	.05	.01	
Intake Rates (I,kilograms/year)				
Critical Individual				(1)
Fish (fresh)	18	18	18	
Shellfish (fresh)	18	--	--	
Water (liters/year)	--(a)	440	440	
Population Average				(77)
Fish (fresh)	3.9	5.1	2.0	
Fish (processed)	2.8	2.6	2.9	
Shellfish	1.8	1.2	0.5	
Water (liters/year)	--(a)	365	365	

(a) The fresh water supply at the seacoast site is not affected by plant effluents.

Other		Reference
Dilution Flow (V liters/year)		
Once-through cooling	1.27×10^{12}	
Cooling tower blowdown	3.47×10^{10}	
Preparation Loss Factor for Seafood (F)	0.8	(54)
Time Between Discharge and Consumption (τ, hours)		
Critical Individual		
Water	24	
Fish	24	
Shellfish	24	
Average Individual in Population		
Water	36	
Fish (fresh)	36	
Fish (processed)	30 days	
Shellfish	30 days	

Evaluation of External Whole Body Doses from Gaseous Effluents

Radiation doses from airborne effluents were calculated using the AIREM computer code (81). This code provides for a Gaussian or bell-shaped concentration profile in the vertical direction and a uniform concentration distribution in the horizontal cross wind direction. The vertical diffusion is limited to a finite mixing height (82) and the technique of image sources is used to account for reflection from both the mixing layer and ground surface.

The basic diffusion equation used in AIREM is a standard sector-averaged equation (83,84) modified to include radionuclide decay by time of flight:

$$\frac{\chi}{Q'} = \frac{\Psi}{Q} = \left(\frac{2}{\pi}\right)^{1/2} \frac{f \exp\left(-h^2/2\sigma_z^2\right) \exp\left(-\lambda t\right)}{\bar{u}\, \sigma_z\, 2\pi r/n}$$

where:

χ = ground level airborne concentration in Ci/m^3,

Ψ = time integrated ground level concentration-exposure in $Ci\text{-}sec/m^3$,

Q' = source release rate in Ci/sec,

Q = time integrated release in Ci (i.e., total release),

f = fractional wind frequency in a sector,

r = distance from the stack in meters,

h = effective stack height in meters,

n = number of sectors,

$2\pi r/n$ = sector width at distance r in meters,

σ_z = standard deviation of the vertical distribution of an assumed gaussian cloud, in meters,

$\bar{u}$ = average wind speed in the sector in m/s,

λ = decay constant of radionuclide in sec^{-1},

and t = transit time from the stack to distance r, in seconds ($t=r/\bar{u}$).

This equation is solved repeatedly for each radionuclide and stability class within each sector for all downwind distances of interest.

The preceding equation provides the ground level air concentration at a distance r from the release point. This concentration is then used to calculate the radiation dose from inhalation and transpiration (tritium), and the deposited activity on the ground surface which contributes to external whole body exposures and to food intake pathways. The inhalation and transpiration doses are computed directly from the ground level air concentration by the following alogrithm:

$$D = \frac{\chi}{Q'} Q \text{ (DCF)},$$

Where D is the dose rate in mrem/year, (χ/Q') is the atmospheric dispersion factor as computed above (sec/m^3), Q is the annual release rate (curies/year), and DCF is an appropriate dose conversion factor (mrem/year per Ci-sec/m^3) for the radionuclide and exposure mode of interest. The inhalation dose conversion factors for radioiodine will be provided in a subsequent section, for all other radionuclides values from (54) were used.

The activity deposited on the ground surface was computed from the ground level air concentration as follows:

$$\omega = \frac{\chi}{Q'} Q \, v_d$$

Where ω is the deposition rate (Ci/m^2 sec), v_d is the deposition velocity (m/sec), and the other quantities are as defined above. The deposition velocity is an empirical factor which is defined as:

$$v_d = \frac{\mathcal{D}}{\chi t}$$

Where $\mathcal{D}$ is the accumulated deposit (Ci/m^2) and χt is the integrated air concentration over the period of measurement. The airborne concentration is depleted uniformly by the deposited activity using the continuity principal.

The accumulated deposit is given by:

$$\mathcal{D} = \frac{\omega}{\lambda_e} [1 - e^{-\lambda_e t}] ,$$

Where ω is the deposition rate as given above, $\mathcal{D}$ is the deposited activity (Ci/m^2), λ_e is an effective removal rate, and t is the time interval. For deposition onto foliage which leads to an ingestion dose, the effective removal rate is defined as:

$$\lambda_e = \lambda + \frac{\ell n \, 2}{12} ,$$

where λ_e is the effective removal rate constant (days^{-1}), λ is the radiological decay constant (days^{-1}), and the remaining term accounts for physical removal by wash-off, wind, and plant growth of the radionuclide from plant surfaces. This latter process is assumed to have a half-time of 12 days (54). Computation of the external whole body dose from deposited radionuclides is performed assuming a uniform semi-infinite plane source by:

$$D = \mathcal{D} \ (DCF') \ ,$$

where D is the annual dose rate (mrem/year), $\mathcal{D}$ is the accumulated activity deposit (Ci/m^2) and DCF' is the dose conversion factor for a semi-infinite plane source (mrem/year per Ci/m^2). Values for DCF' are taken from (54).

The gamma dose rate at the surface of a receptor at a point x_r, y_r, z_r in space from single energy gamma photons emitted from an elemental volume located at point x,y,z of the radionuclide bearing cloud is (84):

$$d\left[DR\left(x_r, y_r, z_r\right)\right] = K \frac{E}{R^2} \ \chi(x,y,z) \ A \ \frac{\mu_e}{\rho} \ \exp\left(-\mu_a R\right) \ B \ dv$$

Where:

DR = dose rate

E = photon energy (MeV)

A = gamma photon abundance (photons/disintegration)

B = buildup factor

μ_a = linear air attenuation coefficient - m^{-1}

$\frac{\mu_e}{\rho}$ = mass energy absorption coefficient of muscle - cm^2/gm

dv = elemental volume - m^3

$\chi(x,y,z)$ = airborne concentration at point x,y,z

= $(\chi/Q')Q$

K = dimensional constant

R = distance between emitting volume and receptor point

where $$R = (x-x_r)^2 + (y-y_r)^2 + (z-z_r)^2$$

The $\chi(x,y,z)$ is computed from the previous relationship for (χ/Q') except that the vertical concentration profile is considered by substituting $(h-z)^2$ for h^2 for each height z.

Integration of this equation over all space will yield the dose rate at the receptor point due to the gamma emitters in the entire plume. Solution of this equation yields the dose rate from mono-energetic gamma photons at a single point on the ground surface and for a single invariant wind speed, wind direction and dispersion regime. The total dose rate from the entire plume material is found by summation over all meteorological conditions and gamma energies with appropriate weighting factors for frequency of occurrence. This integration is performed by R.E. Cooper's EGAD code (56). The EGAD code considers ground and inversion layer reflections and employs an empirical third-order polynomial expression for the buildup factor B. These relationships are more accurate than the simple one-term uncollided flux approximation, $B = 1 + \mu_e R$.

Risk calculations for external whole body photon exposure are based on the average dose to the body allowing for both self shielding and buildup. The dose calculation is performed in two parts. First, the maximum dose to a differential volume of tissue is calculated using appropriate attenuation and buildup factors for air as indicated above, then the average dose to the body is calculated by means of the dose reciprocity theorem (85). Strictly speaking, this implies that the ratio of the average to maximum dose from a finite cloud is the same as the average to maximum dose from a semi-infinite cloud. This is a good assumption for the cases of interest here since in either case, we are dealing with isotopic angular flux distribution (in 2π geometry) and a quasi-equilibrium distribution of photon energies.

To determine the ratio of average to maximum dose from a semi-infinite cloud, we have used an updated version of Adam's (86) solution to this problem as updated by Russell and Galpin (87) to take advantage of the exact photon scattering calculations published by the MIRD committee (70,71). The latter results are tabulated in terms of absorbed fractions ϕ as a function of photon energy and body mass, a 70 kg mass being used for these calculations.

Population-integrated doses (person-rem) are computed by the following expression:

$$D_p = \frac{1}{2} \sum_{j=1}^{n} P_j \left(D_{j-1} + D_j\right) ,$$

where D_p is the total population-integrated dose, P_j is the enclosed population within sector j and D_{j-1} and D_j are the annual individual dose rates for the exposure mode of interest at the inner and outer radial boundaries of sector j.

Radioiodine Thyroid Dose Computations

The control of airborne radioiodine releases is complicated by the diversity of chemical forms which may coexist under certain conditions. Until recently, it has been assumed that the principal chemical form present in reactor effluents was elemental iodine vapor (I_2) and existing control technology was predicated on this assumption. Recent preliminary unpublished studies conducted by the Atomic Energy Commission have indicated that, at several facilities, the predominant chemical form was not elemental iodine but rather a volatile organic iodide, principally methyl iodide, CH_3I. This apparent change in chemical form may not only affect the efficacy of control techniques and physiochemical transfer characteristics and, consequently, the magnitude of the discharge but also may significantly affect the critical exposure pathway leading to man. Airborne radioiodine discharges can result in radiation exposure to man by four principal pathways: air inhalation, milk consumption, ingestion of leafy vegetables and other produce, and external whole-body exposure from activity deposited on the ground. Except for air inhalation, the remaining pathways depend upon the transfer coefficient between air and vegetation on the ground. This transfer coefficient is termed the "deposition velocity." For iodine in the elemental form, the deposition velocity generally ranges between 0.002 and about 0.1 m/sec, depending upon the fraction absorbed on airborne particulates. Generally, deposition velocities in the range of 0.005 to 0.015 m/sec are considered typical for reactor effluents (60, 61). The corresponding deposition velocity for the organic form, methyl iodide, has not been well characterized, but has been shown to be several orders of magnitude smaller than for the elemental form (61-62). This results in negligible deposition of methyl iodide onto vegetation (grass and leafy vegetables) and the ground surface. Because of this, the principal exposure pathway for methyl iodide releases is likely to be direct inhalation rather than milk ingestion. For the elemental form, milk ingestion is likely to be the controlling pathway for iodine-131 if there are grazing animals (dairy cattle or goats) in the vicinity of the plant. This results from the ability of the cow (or goat) to concentrate the radioiodine by virtue of the large area of grass grazed daily (20-80 m^2/day). The exact ratio of the thyroid dose from milk ingestion to air inhalation depends upon parameters (primarily the mass of the thyroid gland, the ventilation [breathing] rate, and the average milk consumption) associated with the age of the individual involved. The critical receptor is usually taken to be a young child because of a smaller thyroid mass and a greater daily milk consumption than for other age groups. These parameters are presented in Table 38 for the selected age groups.

The thyroid population doses were computed for four age groups in order to account for varying thyroid mass, milk consumption, and radiation sensitivity with age. The age groups used were first year infant

(6-month old typical individual), 1-9 years (4-year old typical individual), 10-19 years (14-year old typical individual), and over 20 years (20-year-old typical adult). Three intake modes were considered: air inhalation, vegetation consumption, and milk ingestion. The external whole-body dose from deposited radioiodines was also computed, but this exposure mode was negligible compared to the inhalation and ingestion pathways. In analyzing the radiation exposure from organic iodides, a deposition factor of 1/1000 that of elemental iodine was used. This is an arbitrary value, as the published values (61, 62) show considerable variation. Although the chosen value is somewhat higher than the best available estimate (62), it is felt that the use of the higher value is justified in view of the uncertainty in the chemical form and the possibility of a change in chemical form due to radiolytic or photodissociation of the methyl iodide in the environment or its attraction to airborne aerosols. Both of these processes could drastically affect the chemical form and, hence, the deposition velocity.

The relationship between the dose [equivalent] rate delivered to the critical organ by a radionuclide and the ambient air concentration of that radionuclide can be expressed as:

$$\dot{DE} = \chi(DICF)P \quad ,$$

where $\dot{DE}$ is the dose [equivalent] rate in millirem per year,

χ is the air concentration of the radionuclide (pCi/m^3),

P is the pathway transfer factor (pCi/year per pCi/m^3) relating the intake rate to the ambient air concentration, and DICF is the dose [equivalent] rate delivered per unit intake rate (mrem/year per pCi/year).

The air concentration, χ, is determined by the dispersion models discussed in the preceding section. The pathway transfer factor, P, which relates the intake rate by an individual to the ambient air concentration, will be discussed separately for each pathway in subsequent sections.

The dose [equivalent] -intake conversion factor, DICF, has the same relationship as given in the previous section dealing with water and fish ingestion doses. The parameters for the radioiodines which determine the DICF for ingestion and inhalation are shown in Table 37. It is assumed that the fraction of radioiodine reaching the thyroid gland for milk and vegetation ingestion is identical to that for water ingestion (f_w). The resulting values for DICF for intake by ingestion and inhalation are presented in Table 38 as a function of the age of the receptor.

The Evaluation of Thyroid Dose from Radioiodine Inhalation

The evaluation of the thyroid dose from inhalation is relatively simple as the pathway transfer coefficient, P, is just the breathing rate in m^3/year. These values are shown for the four age groups in Table 39 and the products of P and the dose [equivalent]-intake conversion factors, DICF, are presented in Table 40. The metabolic properties of methyl iodide and the other organic iodine species were assumed to be the same as for the elemental forms. The critical individual doses were calculated assuming no depletion in the air concentration by deposition prior to inhalation. The population dose calculations, however, did account for loss by deposition prior to inhalation.

Table 36

Population Groups Used for

Radioiodine Dose and Risk Evaluations

Age Group	Age of "Typical" Individual	Percent of Total Population Within Age Group
< 1 year old	6 months	1.79%
1 - 9 years old	4 years	16.47%
10 - 19 years old	14 years	19.57%
> 20 years	standard man	62.17%
		100.00%

Source: National Center for Health Statistics, DHEW.

Table 37

Thyroid Dose Parameters

Age Group (years)	0 - 1	1 - 10	10 - 20	> 20
Age of Typical Individual (years)	0.5	4	14	standard man
Fraction of Iodine Reaching Thyroid				
Ingestion (fw)	.50(64)	.50(64)	.37(64)	.30(65)
Inhalation (fa)	.38[a]	.38[a]	.28[a]	.23(65)
Biological Half-Time (days)	20[b]	29[b]	70[b]	100[b]
Effective Decay Energy ($\bar{E}$, MeV)				
I-131	.18(67,68)	.18(67,68)	.19(67,68)	.19(67,68)
I-133	.40[c]	.40[c]	.42[c]	.42[c]
Thyroid Mass (m,grams)	1.9(66)	2.7(66)	12(66)	20(66)

[a] fa was computed as 75 percent of fw as recommended in table 10, page 53 of reference (65).
[b] Upper range values, especially for younger ages, given in reference (66).
[c] Computed using decay scheme presented in (69) and absorbed fractions from references (70) and (71).

Table 38

Radioiodine Dose Conversion Factors Per Unit Activity Intake

Age Group (years)	Age of "Typical" Individual (years)	DICF (mrem/pCi inhaled)	
Inhalation			
		^{131}I	^{133}I
< 1	0.5	1.5×10^{-2}	4.8×10^{-3}
1 - 9	4	1.2×10^{-2}	3.4×10^{-3}
10-19	14	2.4×10^{-3}	6.0×10^{-4}
> 20	20	1.2×10^{-3}	3.0×10^{-4}
Ingestion		DICF (mrem/pCi ingested)	
< 1	0.5	2.0×10^{-2}	6.3×10^{-3}
1 - 9	4	1.6×10^{-2}	4.5×10^{-3}
10 - 19	14	3.1×10^{-3}	8.0×10^{-4}
> 20	20	1.6×10^{-3}	3.9×10^{-4}

Table 39

PATHWAY TRANSFER COEFFICIENT

PATHWAY: Inhalation

Note: The transfer coefficient for the inhalation pathway is the breathing rate in cubic meters per year.

AGE GROUPING	BREATHING RATE [m^3/yr]
6 month-old	1.15×10^3
4 year-old	3.53×10^3
14 year-old	6.44×10^3
Adult	7.30×10^3

Table 40

PRODUCT OF THE TRANSFER COEFFICIENT WITH THE DOSE EQUIVALENT CONVERSION FACTOR FOR THE INHALATION PATHWAY

[mrem/yr per pCi/m^3 in air]

AGE GROUPING	I-131	I-133
6 month-old	17	5.5
4 year-old	42	12
14 year-old	15	3.9
Adult	8.8	2.2
Maximum Individual (4 year-old)	42	12

Evaluation of Thyroid Ingestion Dose

The dose contribution to the thyroid from radioiodine via the air-grass-milk and air-vegetable pathway was evaluated for the three representative sites described previously. The iodine concentrations on vegetation were calculated from annual sector-averaged air concentrations incorporating cloud depletion. The farms and dairy cows were assumed to be uniformly distributed with respect to distance and direction from the source and the average ground deposition in each radial sector was weighted by the fraction of the total area within 50 miles of the plant it comprised. Sufficient dairy and produce farms were assumed to be located within a 50-mile radius of a facility to provide the total milk and produce consumption for the total enclosed population.

The assumptions of uniform location and sufficient milk and produce production are not strictly correct when applied to a specific reactor site. For the representative seacoast site chosen in this study, the nearest dairy farm is approximately 12 miles from the site and milk production within 50 miles is not sufficient to meet the needs of the enclosed population at that distance. Thus, milk must be imported into the region and the population thyroid dose from that facility alone via the milk pathway would be greatly overestimated by the calculation model employed. However, projections of the growth of the nuclear power industry indicate that by the year 2000, a majority of the U. S. population will reside within tens of miles of at least one nuclear power station and a similar situation will undoubtedly exist with respect to dairy farms. Thus, in the future, milk and other produce leaving the 50-mile locality will contribute to the population dose at another location while foodstuffs entering the region are likely to have been produced in the vicinity of another nuclear power plant. For the purpose of calculating the integrated population dose, it is, therefore, reasonable to assume uniform contamination levels in foodstuffs (providing a sufficient decay period for short-lived radionuclides is incorporated) and that the population dose calculated on the basis of total consumption and production within the 50-mile radius is comparable to that calculated allowing for food transfer between regions of production and consumption.

The assumption of a uniform distribution of the farms within the 50-mile radius was compared with the existing farm locations for several actual sites. These locations were found to vary randomly between sites and generally the number of farms enclosed within different radial distances was found to increase with the enclosed land area.

The daily milk or produce intake rates and dose conversion factors were calculated for each of the three selected age groups and weighted according to the fraction of the total U. S. population in each of these age categories as shown in Table 36. These age-weighted dose conversion factors were used together with the area-weighted iodine concentrations and the site-related population densities to arrive at an integrated population dose.

The pathway transfer factors, P, for ingestion of milk and vegetables were computed from the following relationship:

$$P\left(\frac{\text{pCi/day}}{\text{pCi/m}^3}\right) = \frac{v_d \quad RTFI}{\lambda_{eff}}\,[\exp(-\lambda\tau)] \text{ ,}$$

where: v_d is the deposition velocity (m/sec),

R is the initial retention factor,

T is a transfer factor between the deposited activity and the radionuclide concentration in food,

F is an intake modifying factor,

λ_{eff} is the effective removal rate constant from vegetation (seconds^{-1}),

λ is the radioactive decay constant (days)$^{-1}$,

τ is the interval between production (milk) or harvesting (vegetables) and consumption (days), and I is the intake rate of milk or vegetables. The values and units of these parameters together with more complete definitions of R, T, and F for the two pathways are presented in Table 41.

The v_d/λ_{eff} term in this expression times the average air concentration, $\bar{X}$, gives the saturated [equilibrium] ground deposition per unit area. The exponential term accounts for loss by radioactive decay between production and consumption.

The evaluated pathway transfer coefficients, P, for radioiodine-131 and radioiodine-133 are presented in Table 42 for the vegetable ingestion pathway and in Table 43 for the milk ingestion pathway. In both cases, these values are presented separately for each age group. The products of the pathway transfer factors and the dose [equivalent] conversion

Table 41

Parameters Used for the Calculation of Radioiodine Intakes

Parameter	Vegetable Ingestion	Milk Ingestion	Reference
Deposition Velocity (v_d)	meters/second	meters/second	
elemental iodine	0.01	0.01	(76)
organic iodides	1×10^{-5}	1×10^{-5}	(see text)
Initial Retention Factor (R)	pCi/m^2[plant] 0.25	pCi/m^2[deposited] 0.25	(54)
Removal Half-time Due to Loss by Weathering and Growth	days 12	days 12	(54)
Transfer Factor (relating intake to deposited activity)	kilograms/m^2(T) 1 (78)	pCi/liter per pCi/m^2(RT) I-131 0.2	(72)
		I-133 0.1	(73)
Intake Modifying Factor (F)	Fraction remaining after washing & preparation 0.4	Fraction of year cows graze on pasture 0.5	(54)
Decay Interval Between Productio and Consumption (z)	days	days	
Average individual (population)	7	3	(54)
Critical individual	1	1	(54)
Intake Rates (I)	kilograms/year (54,77)	milliliters/day	(74)
6-month-old	0	500	
4-year -old	13	700	
14-year-old	20	660	
adult	30	230	
critical individual	13 (54,77)	1000 (77)	

Table 42

PATHWAY TRANSFER COEFFICIENT [pCi/yr ingested per pCi/m^3 in air]

PATHWAY: Vegetables

AGE GROUPING	I-131	I-133
Average 6 month-old	0	0
Average 4 year-old	4.09×10^3	3.95
Average 14 year-old	6.55×10^3	6.33
Average Adult	9.83×10^3	9.49
Maximum individual [4 year-old]	6.87×10^3	5.39×10^2

NOTE: Only two figures are assumed to be significant.

Table 43

PATHWAY TRANSFER COEFFICIENT [pCi/yr ingested per pCi/m^3 in air]

PATHWAY: Milk

AGE GROUPING	I-131	I-133
Average 6 month-old	8.41×10^4	7.66×10^2
Average 4 year-old	1.18×10^5	1.07×10^3
Average 14 year-old	1.11×10^5	1.01×10^3
Average Adult	3.88×10^4	3.52×10^2
Maximum individual (6 month-old)	2.01×10^5	7.89×10^3

NOTE: Only two figures are assumed to be significant.

factors from Table 38 are presented for each age group in Table 44

Evaluation of Total Thyroid Dose and Associated Health Risks

The population-integrated dose (in person-rems) is obtained by multiplying the average air concentration within a sector by the number of people enclosed within that sector and by a dose conversion factor for an "average" individual in the population. These dose conversion factors are obtained by summing the age-dependent products of the pathway transfer factors and the dose [equivalent] intake conversion factors for inhalation (Table 40) and the two ingestion pathways (Table 44) and multiplying this value by the fraction of the total U.S. population which is in each of the four age groups (Table 36). These values for each of the four age groups are then summed to give a factor for the "average" individual. These calculations are shown for iodine-131 in Table 45 and for iodine-133 in Table 46.

The risk of health effects (principally thyroid cancer) resulting from radiation exposure of the thyroid appears to be age-dependent. For this reason, the population-integrated dose cannot be directly multiplied by a single dose-to-risk conversion factor. This can be accomplished, however, if the age-dependent risk per unit dose values are weighted by the fraction of the total population dose which is delivered to that age group. The resultant dose-to-risk conversion factor for an "average" individual are relatively independent of the radionuclide (I-131 or I-133) or the chemical form of the radioiodine (elemental or organic) as shown in

Tables 45 and 46. For this reason a value of 56 health risks per million person-rem was used for all integrated population doses. The age-dependent risk values for each age group were obtained from data presented in reference 88.

Table 44

Product of the Pathway Transfer Coefficient with the Dose Equivalent Conversion Factor [mrem/year per pCi/m^3 in air]

MILK PATHWAY

AGE GROUPING	I-131	I-133
Average 6 month-old	1.7×10^3	4.8
Average 4 year-old	1.9×10^3	4.8
Average 14 year-old	3.4×10^2	8.1×10^{-1}
Average Adult	6.2×10^1	1.4×10^{-1}
Maximum individual	4.0×10^3	5.0×10^1

VEGETABLE PATHWAY

AGE GROUPING	I-131	I-133
Average 6 month-old	0	0
Average 4 year-old	6.5×10^1	1.8×10^{-1}
Average 14 year-old	2.0×10^1	5.1×10^{-3}
Average Adult	1.6×10^1	3.7×10^{-3}
Maximum individual	1.1×10^2	2.4

Table 45

Radioiodine Population-Weighted Dose Conversion And Dose-To-Risk Conversion Factors Iodine-131

Age Group Range	Age of Typical Individual (Years)	Percentage of Total Population	Risk per 10^6 man-rem	Pathway	Dose Conversion Factor (mrem/yr per pCi/m^3)	Weighted Dose Factor		Percentage of Total Population Dose		Weighted Risks per 10^6 man-rem to Average Population	
						elemental	organic	elemental	organic	elemental	organic
0-1	0.5	1.79	88	Inhalation	17	.3043	.3043				
				Vegetable	0	0	0				
				Milk	1700	30.43	.0304				
				Sub-total		30.734	.3347	6.29	2.08	5.53	1.83
1-9	4	16.47	50	Inhalation	42	6.917	6.917				
				Vegetable	65	10.705	0.011				
				Milk	1900	312.93	0.313				
				Sub-total		330.553	7.241	67.65	44.97	33.82	22.48
10-19	14	19.57	66	Inhalation	15	2.936	2.936				
				Vegetable	20	3.914	0.004				
				Milk	340	66.538	0.066				
				Sub-total		73.388	3.006	15.02	18.67	9.91	12.32
>20	20	62.17	60	Inhalation	8.8	5.471	5.471				
				Vegetable	16	9.947	.010				
				Milk	62	38.545	.038				
				Sub-total		53.963	5.519	11.04	34.28	6.62	20.57
Average Individual in Population						488.64	16.10	100.0	100.0	55.90	57.21

Note: Only two figures are assumed to be significant.

Table 46

Radioiodine Population-Weighted Dose Conversion And Dose-To-Risk Conversion Factors Iodine-133

Age Group Range	Age of Typical Individual (Years)	Percentage of Total Population	Risk per 10^6 man-rem	Pathway	Weighted Dose Factor (mrem/yr per pCi/m3)		Percentage of Total Population Dose		Weighted Risks per 10^6 man-rem to Average Population	
					Elemental	Organic	elemental	organic	elemental	organic
0-1	0.5	1.79	88	Inhalation	0.0984	0.0984				
				Vegetable	0	0				
				Milk	.0859	<.0001				
				Sub-total	0.1844	0.0984	3.44	2.34	3.03	2.06
1-10	4	16.47	50	Inhalation	1.9760	1.9760				
				Vegetable	.0296	<.0001				
				Milk	.7906	.0008				
				Sub-total	2.797	1.977	52.17	46.99	26.08	23.50
10-20	14	19.57	66	Inhalation	.7632	.7632				
				Vegetable	.0010	<.0001				
				Milk	.1585	.0002				
				Sub-total	.9227	.7635			11.36	11.98
>20	20	62.17	60	Inhalation	1.3680	1.3680				
				Vegetable	.0023	<.0001				
				Milk	.0870	<.0001				
				Sub-total	1.457	1.369	27.18	32.52	16.31	19.51
Average Individual in Population					5.361	4.207	100.0	100.0	56.78	57.04

ESTIMATED RADIATION EXPOSURE FROM NUCLEAR REACTOR EFFLUENTS

Radiation exposure to the population within 50 miles of the reactor site was calculated using the methods outlined in the preceding section. These exposures were determined for each of the postulated treatment options and for each of the three representative site classes described in Table 1. In addition, a hypothetical "average site" was constructed from these three sites by weighting the radiation dose at each site by the projected distribution of sites:

$$D_{\text{"average"}} = 0.25D_{\text{seacoast}} + 0.60D_{\text{river}} + 0.15D_{\text{lake}}$$

Liquid Effluents

It is convenient to distinguish the radiation dose from tritium releases and the dose produced from all other radionuclides discharged in liquid effluents. These doses are shown for tritium in Table 47 and for all other radionuclides in Table 48 . Despite the relatively large contribution to the total activity (curies) discharge due to tritium, its biological significance in terms of radiation exposure to local population groups is very small. This disproportionate significance results from the radiological parameters associated with tritium, primarily the low average energy of its beta emission (6 keV compared to 1131 keV for the combined decay of strontium-90 and its daughter yttrium-90), its relatively short residence time in the body (12 days) resulting from its incorporation into body water, and its generally uniform distribution throughout the body instead of being localized in one organ.

Table 47

Dose Equivalents from Tritium Releases

System	Population Dose Equivalents (man-rem/yr)						Individual Dose Equivalents (mrem/yr)			Annual Discharge (Ci/yr)
	Whole Body			Thyroid			Whole Body			
	Seacoast	River	Lake	Seacoast	River	Lake	Seacoast [a]	River	Lake	
BWR-1,2,& 3	.0015	.12	.02	.0015	.12	.02	.00012	.001	.001	200
BWR-4	.00095	.08	.013	.00095	.08	.013	.000079	.00073	.00073	130
PWR-1,2	.0087	.74	.12	.0087	.74	.12	.00074	.0067	.0067	1200
PWR-3,4	.0055	.47	.076	.0055	.47	.076	.00047	.0042	.0042	760

[a] Does not include shellfish.

Table 48

SUMMARY TABLE: LIQUID RADIOACTIVE WASTE SYSTEMS, ESTIMATED COSTS, AND DOSE EQUIVALENTS

System	Estimated Capital Cost	Estimated Annual Cost	Dose Equivalents From Non-Tritium Radioactive Releases												Annual Health Effects At Average Site		
			Individual Whole Body Dose Equivalents (mrem/yr)				Population Dose Equivalents (person-rem/yr)										
							Whole Body				Thyroid						
			Seacoast	River	Lake	Average Site	Seacoast	River	Lake	Average Site	Seacoast	River	Lake	Average Site	Whole Body	Thyroid	Total
BWR Systems[a]																	
BWR-1	416,000	136,500	2.2	24	4	[illegible]	570	[illegible]	141	1250	2200	13,500	2100	9120	2.28	0.511	2.79
BWR-2	1,726,000	401,000	0.019	[illegible]	0.14	[illegible]	1.48	14.1	46	8.92	22	135	31	91.2	0.0062	0.0051	.011
BWR-3	2,314,000	560,000	0.0032	0.66	[illegible]	[illegible]	0.48	[illegible]	[illegible]	4.23	9.9	60.9	14	41.1	0.0030	0.0023	.0053
BWR-4	3,271,000	680,000	0.0001	[illegible]	[illegible]	[illegible]	0.16	[illegible]	[illegible]	0.038	0.96	6.0	1.4	4.10	2.7(-5)[b]	2.3(-4)	2.6(-4)
PWR Systems[a]																	
PWR-1	254,000	57,000	6.0	190	190	144	1060	20,200	562	12,300	3100	18,700	4200	12,600	8.61	0.11	9.22
PWR-2	509,000	121,000	0.11	2.9	2.9	2.20	14.8	190	7.84	179	150	914	210	617	0.13	0.035	0.165
PWR-3	1,213,000	285,000	0.013	0.26	0.06	0.027	1.74	9.4	1.76	47.3	1.6	9.58	2.2	6.48	0.033	0.00036	0.033
PWR-4	1,517,000	429,000	0.0001	[illegible]	0.0005	[illegible]	0.0020	0.021	[illegible]	0.017	0.41	2.48	0.52	1.68	[illegible]	9.4(-5)	0.0001

[a] Values are for a 2-unit, 1000 MW(e) each, power station site.
[b] Expressions such as 2.7 (-5) mean 2.7 x 10^{-5}, or 0.000027.

For light-water nuclear power reactors there are only two important pathways for tritium exposure: ingestion in food and drinking water. The other principal radionuclides, however, produce radiation exposure via several exposure modes. As the effluent composition changes with different treatment options, the relative contributions from these different pathways also changes. For the pressurized-water reactors, (PWRs), the only radionuclides which produce significant radiation exposure are tritium and the two cesium isotopes: cesium-134 and cesium-137. These latter two radionuclides comprise over 95% of the total whole-body dose from PWR cases 1 and 2. For ingestion of drinking water and fish, the cesium-134 delivers about 60% of the dose and the cesium-137 about 33%. Tritium becomes significant in PWR case 3, accounting for most of the radiation exposure from drinking water ingestion, but the cesiums are still significant via fish consumption due to the high biological reconcentration factor exhibited by freshwater fish. Only in PWR case 4 does tritium become the largest source of exposure, the cesiums accounting for only 25% of the dose from fish consumption.

For the boiling-water reactors (BWRs) other radionuclides become significant in addition to tritium and the two cesiums. Because of the lower amount of tritium discharged from boiling-water reactors (200 curies versus 800 from the PWR station), it does not constitute a significant source of radiation dose until the majority of other radionuclides are removed in BWR treatment options 3 and 4. In BWR cases 1 and 2, the dose contribution from fish consumption is due to the two cesiums, with cesium-134 contributing over 60% of the dose as a consequence of the greater cesium-134 activity released. For drinking water intake a variety of radionuclides contribute to the radiation dose, the most important being the two cesiums and strontium-90. These three radionuclides comprise 60% of the radiation dose delivered via this intake pathway.

In BWR cases 3 and 4 tritium contributes significantly to the dose from drinking water intake, accounting for about half of the dose for case 3 and essentially all of the dose in case 4 from drinking water. The tritium contribution from fish intake in case 4 is about 15% of the total, the two cesiums contributing the remainder of the dose.

Radiation doses from present liquid effluent control systems are relatively small compared to other effluent paths. Present treatment systems (BWR-2 and PWR-2) result in individual whole body dose equivalents of between 0.1 and 2.2 mrem/year and whole-body population-integrated dose equivalents between 9 and 180 person-rem. The radioiodine discharges produce thyroid population dose equivalents between 160 and 1100 person-rem, the larger value associated with PWR effluents.

Table 48 shows that the dose reduction factor for liquid effluent treatment systems can reach approximately 1,000,000 for the cases PWR-4 or BWR-4 as compared to the hypothetical zero treatment systems (BWR-1 and PWR-1). In all cases, the river site produces the maximum dose as a consequence of the smaller dilution factor afforded at this site. The population doses however, differ appreciably between the two reactor types and varies by an order of magnitude or more in some cases. The dose variation between BWRs and PWRs is due partly to the different composition of the effluent stream.

Noble Gases

The radiation exposure from pressurized water reactor gaseous effluents is relatively low due to the small quantity of undecayed noble gas radionuclides emitted. These releases are much smaller from the PWR than from an equivalent boiling-water reactor, as indicated in Tables 49 and 50. This is a consequence of the presence of a secondary coolant loop in the pressurized-water reactor. In the BWR, the radiolytic hydrogen and oxygen and other air ejector effluents constitute a significant volume. In pressurized water reactors, the secondary coolant system isolates the condenser air ejector from the radiolytic gases in the primary coolant and, consequently, yields significantly smaller off-gas volumes. This permits tank storage with the resultant decay of short-lived gaseous radionuclides.

The integrated population dose equivalents within 50 miles at the three representative sites and an "average" site are shown in Table 49 for a pressurized water reactor. As can be seen, the population integrated dose equivalents are small, even for the hypothetical zero treatment case Class-0 and the great majority of the dose results from the 5.3-day xenon-133 for all treatment options. This is a consequence of its relatively long half-life compared to the other noble gases. Although krypton-85 has a significantly longer half-life (10.8 years), its production in fission is about a factor of twenty lower than xenon-133 and it emits only a low energy beta particle. The comparable contribution of krypton-85 is about 0.006 person-rem/yr within fifty miles of the reactor. Due to the long half-life of krypton-85, its release is not greatly affected by holdup of the offgas as is xenon-133, and only true separative noble gas control systems such as cryogenic distillation or selective absorption modify the release of krypton-85.

Table 49

SUMMARY TABLE: PWR NOBLE GAS DOSE EQUIVALENTS AND HEALTH EFFECTS (2 units, 1000 MW(e) each)

Discharge Control Option	Class (system designation)	Estimated Capital Cost	Estimated Annual Cost	Maximum Individual Whole Body Dose Equivalent Rate (mrem/yr)								Whole Body Population Dose Equivalent Rate (person-rem/yr)				Annual Health Effects			
				Site Boundary				Nearest Resident											
				Seacoast	River	Lake	Average Site	Seacoast	River	Lake	Average Site	Seacoast	River	Lake	Average Site[a]	Seacoast	River	Lake	Average Site[a]
Source Term	0	$ 0	$ 0	1.3	6.1	6.0	4.9	1.4	3.7	6.0	3.5	15.4	125	10.2	89.1	0.0108	0.0878	0.0071	0.0624
Charcoal Adsorption (15D XE)	1A-15	$ 360,000	$ 60,000	0.14	0.66	0.65	0.53	0.16	0.40	0.65	0.38	1.8	15.5	1.2	18.3	0.0012	0.0108	0.0008	0.0128
Charcoal Adsorption (30D XE)	1A-30	$ 540,000	$ 90,000	0.04	0.17	0.17	0.14	0.04	0.10	0.17	0.10	0.4	3.8	0.3	10.8	0.0003	0.0026	0.0002	0.0076
Charcoal Adsorption (45D XE)	1A-45	$ 720,000	$120,000	0.02	0.10	0.10	0.08	0.02	0.06	0.10	0.06	0.3	2.2	0.2	9.8	0.0002	0.0015	0.0001	0.0069
Charcoal Adsorption (60D XE)	1A-60	$ 900,000	$150,000	0.02	0.09	0.09	0.08	0.02	0.06	0.09	0.05	0.2	2.0	0.2	9.7	0.0002	0.0014	0.0001	0.0068
Physical Holdup (15D)	1B-15	$ 500,000	$180,000	0.14	0.65	0.65	0.53	0.15	0.40	0.65	0.37	1.7	15.4	1.2	18.2	0.0012	0.0108	0.0008	0.0128
Physical Holdup (30D)	1B-30	$ 600,000	$190,000	0.04	0.17	0.17	0.14	0.04	0.10	0.17	0.10	0.4	3.8	0.3	10.8	0.0003	0.0026	0.0002	0.0076
Physical Holdup (45D)	1B-45	$ 700,000	$230,000	0.02	0.10	0.10	0.08	0.02	0.06	0.10	0.06	0.3	2.2	0.2	9.8	0.0002	0.0015	0.0001	0.0069
Physical Holdup (60D)	1B-60	$ 800,000	$250,000	0.02	0.09	0.09	0.08	0.02	0.06	0.09	0.05	0.2	2.0	0.2	9.7	0.0002	0.0014	0.0001	0.0068
Cryo. Dist. or Sel. Absn. (XE DF=1000)	2A	$1,500,000	$600,000	0.02	0.10	0.10	0.08	0.02	0.06	0.10	0.06	0.3	2.1	0.2	1.6	0.0002	0.0014	0.0001	0.0011
Cryo. Dist. or Sel. Absn. (XE DF=10,000)	2B	$1,500,000	$600,000	0.02	0.09	0.09	0.07	0.02	0.06	0.09	0.05	0.2	1.9	0.2	1.5	0.0002	0.0014	0.0001	0.0010
Classes 1A-15 & 2A	3	$1,860,000	$660,000	0.02	0.09	0.09	0.07	0.02	0.06	0.09	0.05	0.2	1.9	0.2	1.6	0.0002	0.0013	0.0001	0.0011
Cover Gas Recycle	4	$2,000,000	$580,000	0.01	0.04	0.04	0.03	0.01	0.02	0.04	0.02	0.1	0.6	0.1	0.4	0.0001	0.0004	<0.0001	0.0003
SJAE Char. Adsn. plus Class 1A-15	5A	$1,260,000	$210,000	0.14	0.65	0.64	0.52	0.15	0.39	0.65	0.37	1.7	15.3	1.2	18.2	0.0012	0.0137	0.0008	0.0128
SJAE Char. Adsn. plus Class 2A	5B	$2,400,000	$750,000	0.02	0.08	0.08	0.07	0.02	0.05	0.08	0.05	0.2	1.8	0.1	1.4	0.0002	0.0013	0.0001	0.0010
SJAE Char. Adsn. plus Class 3	5C	$2,760,000	$810,000	0.02	0.08	0.08	0.06	0.02	0.06	0.08	0.04	0.2	1.6	0.1	1.3	0.0001	0.0012	0.0001	0.0009
SJAE Char. Adsn. plus Class 4	5D	$2,900,000	$730,000	0.01	0.03	0.03	0.02	0.01	0.02	0.03	0.02	0.1	0.4	<0.1	0.3	<0.0001	0.0003	<0.0001	0.0002

[a] Includes worldwide krypton-85 commitment.

As shown in Table 49 the application of cryogenic distillation or selective absorption (Class 2A) results in virtually the same individual and population doses as the use of 60-day xenon holdup (Class 1A-60). These two classes of discharge control options are very effective in minimizing primary gas releases, so much so that secondary sources of release now dominate the sum of all releases from all pathways. To illustrate this, a class of discharge control options was devised such that 15-day xenon holdup (Class 1A-15) was followed by selective absorption for cryogenic distillation (Class 2A). This combination is shown as Class 3 and results in only slightly reduced population and individual dose equivalents (see table 37).

One last discharge control option considered for primary gases is the cover gas recycle system. Because this option reduces the primary coolant concentration of longer-lived (i.e., >12 hr) radionuclides, the release of these nuclides from all pathways is reduced, resulting in a further factor of two reduction in individual dose equivalents but a factor of four reduction in population doses.

Finally, one option for treating a secondary source of noble gases was considered, namely, the use of charcoal adsorption beds on the air ejector (Classes 5A through 5D). This option is only effective for those cases in which secondary sources dominate the total release (Classes 5B, 5C, and 5D). Individual doses and whole body population doses are reduced slightly in each case.

As previously discussed, the differences in design between BWRs and PWRs allows greater release rates from the BWR, especially in the case of shorter-lived radionuclides. As shown in the summary table, Table 50, a BWR of the

Table 50

SUMMARY TABLE: BWR NOBLE GAS DOSE EQUIVALENTS AND HEALTH EFFECTS (2 UNITS, 1000 MW(E) EACH)

Discharge Control Option	Class (system designation)	Estimated Capital Cost	Estimated Annual Cost	Maximum Individual Whole Body Dose Equivalent Rate (mrem/yr)[a]								Whole Body Population Dose Equivalent Rate (person rem/yr)[a]				Annual Health Effects			
				Site Boundary				Nearest Resident											
				Seacoast	River	Lake	Average Site	Seacoast	River	Lake	Average Site	Seacoast	River	Lake	Average Site	Seacoast	River	Lake	Average Site[b]
Source (30 min delay, 100 m stack)	0	$3,200,000	$ 600,000	136	321	222	253	72.8	214	222	180	1030	4810	521	3220	0.72	3.37	0.365	2.25
Recombiner: Holdup (24h) 100 m stack	1	$6,500,000	$1,400,000	2.5	7.8	5.3	6.1	1.8	5.1	5.3	4.3	35.8	207	18.6	140	0.0250	0.145	0.0130	0.0981
Charcoal Adsorption (10 DXE)	2-10	$5,600,000	$1,200,000	2.7	11.8	11.6	9.5	2.8	7.2	11.6	6.7	22.5	146	14.2	99.4	0.0157	0.102	0.0100	0.0696
Charcoal Adsorption (20 DXE)	2-20	$6,000,000	$1,330,000	1.1	4.8	4.8	3.9	1.2	2.7	4.8	2.7	5.1	34.7	3.5	26.6	0.0036	0.0243	0.0025	0.0186
Charcoal Adsorption (40 DXE)	2-40	$7,000,000	$1,620,000	1.0	4.3	4.3	3.4	1.1	2.4	4.3	2.4	3.8	24.0	2.6	19.8	0.0027	0.0168	0.0018	0.0138
Charcoal Adsorption (60 DXE)	2-60	$8,000,000	$1,910,000	1.0	4.2	4.3	3.4	1.1	2.4	4.3	2.4	3.8	23.4	2.6	19.4	0.0026	0.0164	0.0018	0.0136
Cryo. Dist. or Sel. Absn. (XE DF=1000)	3A	$7,000,000	$1,400,000	1.4	6.0	6.0	4.8	1.5	3.5	6.0	3.4	7.3	44.7	4.8	29.4	0.0051	0.0313	0.0034	0.0206
Cryo. Dist. or Sel. Absn. (XE DF=10,000)	3B	$7,000,000	1,400,000	1.0	4.4	4.4	3.6	1.1	2.5	4.4	2.4	4.1	25.3	2.8	16.6	0.0029	0.0177	0.0020	0.0116
Classes 2-10 & 3A	4	$12,600,000	$2,600,000	1.0	4.2	4.3	3.4	1.1	2.4	4.3	2.4	3.8	23.4	2.6	15.4	0.0026	0.0164	0.0018	0.0108
Clean Steam Plus Class 1	5A	$7,500,000	$1,600,000	1.9	6.1	4.0	4.7	1.4	4.0	4.0	3.4	32.6	193	17	131	0.0229	0.1350	0.0119	0.0908
Clean Steam Plus Class 2-10	5B	$6,600,000	$1,400,000	1.8	8.2	7.9	6.5	1.9	5.0	7.9	4.7	19.3	127	12.1	86.8	0.0135	0.0889	0.0084	0.0608
Clean Steam Plus Class 2-60	5C	$9,000,000	$2,110,000	0.1	0.5	0.5	0.4	0.1	0.3	0.5	0.3	0.5	3.6	0.3	6.4	0.0004	0.0026	0.0002	0.0045
Clean Steam Plus Class 3A	5D	$8,000,000	$1,600,000	0.5	2.3	2.3	1.9	0.6	1.4	2.3	1.3	4.1	24.7	2.6	16.2	0.0028	0.0173	0.0018	0.0114
Clean Steam Plus Class 4	5E	$13,600,000	$2,800,000	0.1	0.5	0.5	0.4	0.1	0.3	0.5	0.3	0.5	3.7	0.3	2.4	0.0004	0.0026	0.0002	0.0017

[a] Dose equivalents and health effects from noble gases only. Radioiodines are also removed from the air ejector effluent and contribute 1470-3710 person-rem/yr and 0.083-0.21 health effects/yr at the average site (where the low estimate assumes organic radioiodine and the high estimate assumes elemental radioiodine) for all discharge control options except Class 0.

[b] Includes worldwide krypton-85 commitment.

1,000 MWe size with the control system typical of BWRs presently operating would deliver site boundary whole body dose equivalents of over one hundred millirem. Population dose equivalents might be expected to reach over one thousand person-rem/yr at an average site with one 1,000 MWe unit. Furthermore, the release of radioiodine would account for site boundary individual dose equivalents of many hundred millirem and thyroid population dose equivalents of almost two thousand thyroid person-rem at an average site with one unit (see Table 52, page 139).

In selecting the discharge control options, consideration was given to those options in current use as well as to those presently committed. Classes of options were arbitrarily defined. Unlike PWRs, dose equivalents are not dominated as strongly by xenon-133.

The first option considered was the use of physical holdup to allow decay of shorter-lived radionuclides. This system is similar in concept to the typical PWR gas decay tank system. However, long decay times (such as 60 days) cannot be routinely obtained because of the high air ejector offgas flow rates, even after the use of a recombiner to eliminate the condensible gases. Such a system is limited by the condenser air in-leakage flow rate,and delay time is inversely proportional to such in-leakage. Table 50 shows that for a nominal 24-hour delay, individual whole body dose equivalents are lowered to less than five millirem and population whole body dose equivalents are reduced to less than 100 person-rem for a 1,000 MWe BWR at an average site. However, radioiodine dose equivalents are less than one per cent of those for the source term, Class 0, discussed above.

The second discharge control option considered was the use of a recombiner followed by ambient temperature charcoal adsorption beds. As shown in Table 50 for Classes 2-10 through 2-60, a choice exists as to the number of beds and hence the delay achieved. Class 2-10 fails to quite achieve the individual whole body dose equivalents of the 24 hour holdup system, Class 1. This is due to the differing delay times of xenon and krypton in the charcoal beds; although a 10-day xenon decay is achieved, the kryptons are delayed only one-fifteenth as long, allowing more of the radiologically significant short-lived krypton isotopes to be released. Population dose equivalents, however, are lower than Class 1 as the xenon isotopes are delayed effectively enough to offset the increase in population dose equivalent achieved by krypton-88, now released without a 100 m stack. Class 2-20 is capable of reducing both the individual and population whole body dose equivalents to well below those of Class 1. Only slight reductions in dose equivalents are achieved by the addition of more charcoal adsorber beds. It should be pointed out that because of the recombiner condensers and the huge amounts of charcoal employed, radioiodine is also removed from the air ejector effluent.

Cryogenic distillation or selective absorption, the third discharge control option selected, appears to be about as effective as the charcoal adsorption systems providing 20- to 40-day xenon delay. However, performance projections are more uncertain and thus the evaluation for two different xenon DFs, Classes 3A and 3B. Radioiodine is also effectively removed from the air ejector offgas.

Combining a charcoal adsorption system and a cryogenic distillation system (or selective absorption), the fourth discharge control option selected, provides negligible reductions in dose equivalents over the use of either alone compared to the incremental cost incurred.

Finally, one option for treating a secondary source of noble gases was considered, namely, the application of a clean steam source to the turbine gland seal. This option is more effective for those cases in which secondary sources dominate the total releases. Reduction factors of from 1.3 to 10.0 in the individual whole body dose equivalents are possible but whole body population dose equivalents are reduced by no more than a factor of three.

Radioiodine

The radiation doses from PWR radioiodine releases are shown in Table 51 for both the general population integrated dose and the dose to the maximum individual at the site boundary. Table 51 shows that, with the release conditions postulated in preceding sections, the PWR site boundary thyroid dose equivalent can be reduced to 10-22 mrem/year for radioiodine at an average site (the lower dose being due to the smaller contribution from milk for the organic iodines). However, the assumption of an infant spending 100% residence at the site boundary is not realistic, as the nearest residence and dairy farm are not usually located at the same position nor at the site boundary. Dose equivalents at all three locations are given in Table 51 to show the reduction that may be obtained.

The radioiodine doses from BWR effluents are presented in Table 52. The prohibitively high site boundary doses shown for a release from the 30-minute holdup system are purely hypothetical. All current plants employing this system are much smaller and are planning to retrofit augmented systems which will effectively minimize radioiodine from the

Table 51

SUMMARY TABLE. PWR RADIOIODINE DOSE EQUIVALENTS AND HEALTH EFFECTS (2 units, 1000 MW(e) each)

Control Option Added	System Designation	Estimated Capital Cost (cumulative)	Estimated Annual Cost (cumulative)	Radioiodine Release (Ci/yr)		Maximum Individual Thyroid Dose Equivalent Rate (mrem/yr) Site Boundary				Maximum Individual Thyroid Dose Equivalent Rate (mrem/yr) Nearest Residence				Maximum Individual Thyroid Dose Equivalent Rate (mrem/yr) Farm				Population Thyroid Dose Equivalent Rate (person-rem/yr)				Annual Health Effects
				I-131	I-133	Seacoast	River	Lake	Average Site	Seacoast	River	Lake	Average Site	Seacoast	River	Lake	Average Site	Seacoast	River	Lake	Average Site	Average Site
Elemental Cases																						
[illegible]	[illegible]	$ 0	$ 0	[illegible]	[illegible]	112	96	337	449	116	325	337	270	6.5	28.6	27.3	27.9	50.4	296	32	196	[illegible]
[illegible]	[illegible]	$ 700,000	$ 120,000	[illegible]	[illegible]	167	393	233	311	81	225	233	190	4.5	19.8	18.9	15.9	34.9	205	22	136	[illegible]
[illegible]	[illegible]	$ 900,000	$ 160,000	[illegible]	[illegible]	74	192	108	144	[illegible]	104	[illegible]	87.9	2.0	9.7	8.7	[illegible]	16.2	95.1	10.2	62.6	[illegible]
[illegible]	[illegible]	$2,950,000	$ 460,000	[illegible]	[illegible]	21.0	65.6	49.1	52.1	[illegible]	37.7	[illegible]	31.9	0.7	3.3	3.2	2.6	5.9	34.4	3.7	22.3	[illegible]
[illegible]	[illegible]	[illegible]	$ 560,000	[illegible]	[illegible]	13.1	31.9	19.0	22.8	[illegible]	18.7	19.0	15.5	0.4	1.6	1.6	1.3	2.8	16.7	1.8	11.0	[illegible]
[illegible]	[illegible]	[illegible]	$ 860,000	[illegible]	[illegible]	8.3	25.1	12.1	16.7	4.2	11.6	12.1	9.8	0.2	1.0	1.0	0.8	1.8	10.6	1.1	7.0	[illegible]
[illegible]	[illegible]	[illegible]	$1,160,000	[illegible]	[illegible]	3.8	8.1	4.0	6.6	[illegible]	4.7	4.5	4.0	0.1	0.4	0.4	0.3	0.7	4.3	0.5	2.9	[illegible]
[illegible]	[illegible]	[illegible]	$1,260,000	[illegible]	[illegible]	[illegible]	7.9	4.7	8.2	1.6	4.5	4.7	3.8	0.1	0.4	0.4	0.3	0.7	4.1	0.4	2.7	[illegible]
Organic Cases																						
[illegible]	[illegible]	$ 0	$ 0	[illegible]	[illegible]	[illegible]	191	116	152	39.6	111	116	93.8	2.2	9.7	9.3	7.8	[illegible]	313	34.3	210	[illegible]
[illegible]	[illegible]	$ 700,000	$ 120,000	36.4	78.1	71.1	174	101	138	35.7	96.5	101	84.3	2.0	8.8	8.4	7.0	[illegible]	208	30.9	185	[illegible]
[illegible]	[illegible]	$ 950,000	$ 160,000	25.2	22.1	50.7	176	73.4	98.1	25.4	71.2	71.4	60.7	1.4	6.3	6.0	5.0	34.1	204	21.0	132	[illegible]
[illegible]	[illegible]	[illegible]	$ 760,000	16.1	16.1	17.8	82.1	44.0	60.2	16.9	47.1	44.0	40.0	5.9	4.2	4.0	3.3	23.1	138	14.6	89.5	[illegible]
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	5.1	12.6	7.6	10.0	2.6	7.2	7.6	6.1	0.1	0.6	0.6	0.5	[illegible]	20.1	2.2	13.6	[illegible]
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	3.3	5.5	3.3	3.8	1.1	3.2	3.3	2.1	0.1	0.3	0.3	[illegible]	1.6	9.2	1.0	6.1	[illegible]
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	.38	2.2	5.4	3.2	4.1	1.1	3.1	3.2	2.0	[illegible]	0.3	0.3	0.3	1.5	9.0	1.0	5.9	[illegible]
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	2.2	5.4	3.2	4.1	1.1	3.1	3.2	2.6	0.1	0.3	0.3	0.3	[illegible]	8.9	1.0	5.5	[illegible]

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[a] An expression such as 1.1 (-2) is equivalent to 0.011.

[b] Containment purge is also routed through this filter.

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Table 52

SUMMARY TABLE: BWR RADIOIODINE DOSE EQUIVALENTS AND HEALTH EFFECTS (2 units, 1000 MW(e) each)

Control Option Added	System Designation	Estimated Capital Cost (cumulative)	Estimated Annual Cost (cumulative)	Radioiodine Release (Ci/yr)		Maximum Individual Thyroid Dose Equivalent Rate (mrem/yr) Site Boundary				Maximum Individual Thyroid Dose Equivalent Rate (mrem/yr) Nearest Residence				Maximum Individual Thyroid Dose Equivalent Rate (mrem/yr) Farm				Population Thyroid Dose Equivalent Rate (person - rem/yr)				Annual Health Effects
				I-131	I-133	Seacoast	River	Lake	Average Site	Seacoast	River	Lake	Average Site	Seacoast	River	Lake	Average Site	Seacoast	River	Lake	Average Site	Average Site
Elemental Cases																						
Presently Operating[a]	BGIE 1A	($1 200 000)	($ 600 000)	(30.0)	(176)	(150)	(1370)	(112)	(876)	(112)	(925)	(112)	(606)	(72)	(95.8)	(115)	(92.7)	(1130)	(5800)	(596)	(3850)	2.2(-1)[b]
None[c]	BGIE 1	$ 0	$ 0	1.02	5.84	168	411	244	325	84	236	244	199	4.7	20.7	19.8	16.6	36.0	212	22.7	139	7.8(-3)
Clean Steam Valves > 2.5" Diameter	BGIE 2	$1 800 000	$ 300 000	0.224	1.28	36.9	90.2	53.6	71.4	18	51.7	53.6	43.7	1.0	4.6	4.3	3.7	7.9	46.5	5.0	30.6	1.7(-3)
Turbine Bldg Charcoal Adsorber	BGIE 3	$4 300 000	$ 750 000	0.044	0.250	7.2	17.7	10.5	14.0	3.6	10.1	10.5	8.5	0.2	0.9	0.9	0.7	1.6	9.1	1.0	6.0	3.4(-4)
Upgrade to Deep Bed Charcoal Adsorber, Turbine Bldg	BGIE 4	$5 100 000	$1 200 000	0.026	0.147	4.3	10.5	6.2	8.3	2.1	6.0	6.2	5.1	0.1	0.5	0.5	0.4	0.9	5.4	0.6	3.6	2.0(-4)
Reactor Bldg Charcoal Adsorber	BGIE 5	$7 100 000	$1 600 000	0.017	0.058	2.7	6.6	3.9	5.2	1.4	3.8	3.9	3.2	0.1	0.3	0.3	0.3	0.6	3.4	0.4	2.3	1.3(-4)
Radwaste Bldg Charcoal Adsorber	BGIE 6	$7 350 000	$1 640 000	0.010	0.055	1.6	4.0	2.4	3.2	0.8	2.3	2.4	1.9	0.0	0.2	0.2	0.2	0.4	2.1	0.2	1.4	7.6(-5)
Turbine Gland Seal Clean Steam	BGIE 7	$7 950 000	$1 840 000	0.004	0.022	0.7	1.6	0.9	1.3	0.3	0.9	0.9	0	0.0	0.1	0.1	0.1	0.1	0.8	0.1	0.5	3.1(-5)
Organic Cases																						
Presently Operating[a]	BGIO 1A	($1 200 000)	($ 600 000)	(178)	(955)	(21.8)	(199)	(16.2)	(127)	(19.2)	(134)	(16.2)	(87.6)	(10.5)	(13.9)	(16.7)	(13.5)	(503)	(2570)	(285)	(1710)	9.6(-2)
None[c]	BGIO 1	$ 0	$ 0	28.1	105	93.2	227	135	180	46.6	130	135	110	2.6	11.5	10.9	9.2	61.5	362	38.9	238	1.3(-2)
Reactor Bldg Charcoal Adsorber	BGIO 2	$2 000 000	$ 400 000	2.85	16.3	12.0	29.3	17.4	23.2	6.0	16.8	17.4	14.7	0.3	1.5	1.4	1.2	7.8	46.0	4.9	30.3	1.7(-3)
Upgrade to Deep Bed Charcoal Adsorber, Reactor Bldg	BGIO 3	$2 500 000	$ 800 000	1.30	7.40	5.4	13.1	7.9	10.5	2.7	7.6	7.9	6.4	0.1	0.7	0.6	0.5	3.6	20.9	2.2	13.6	7.7(-4)
Radwaste Bldg Charcoal Adsorber	BGIO 4	$2 750 000	$ 840 000	1.19	6.79	5.0	12.2	7.2	9.7	2.5	7.0	7.2	5.9	0.1	0.6	0.6	0.5	3.3	19.2	2.1	12.6	7.1(-4)
Clean Steam Valves > 2.5" Diameter	BGIO 5	$4 550 000	$1 140 000	0.39	2.23	1.6	4.0	2.4	3.2	0.8	2.3	2.4	1.9	0.0	0.2	0.2	0.2	1.1	6.3	0.7	4.1	2.3(-4)
Turbine Bldg Charcoal Adsorber	BGIO 6	$7 050 000	$1 590 000	0.21	1.21	0.9	2.2	1.3	1.7	0.4	1.2	1.3	1.0	0.0	0.1	0.1	0.1	0.6	3.4	0.4	2.2	1.3(-4)
Turbine Gland Seal Clean Steam	BGIO 7	$7 650 000	$1 790 000	0.204	1.17	0.9	2.1	1.2	1.7	0.4	1.2	1.2	1.0	0.0	0.1	0.1	0.1	0.6	3.3	0.4	2.2	1.2(-4)

[a] Costs reflect that of 30 minute air ejector delay line, 100 m stack for two units, and 2 minute turbine gland seal delay line and have been taken into account for the noble gases. Values for other parameters include secondary release pathway contributions as well as air ejector contribution.

[b] Expressions such as 2.2 (-1) are equivalent to 0.22.

[c] These and subsequent values are appropriate to secondary radioiodine release pathways only, as would be the case for an augment noble gas treatment system that also limits air ejector radioiodine releases to extremely low levels (<0.001 Ci/yr).

air ejector offgas, which is the largest source of radioiodine. Thus, the "source term" for future BWRs will consist of only secondary release pathways, case BGIE-1 or BGIO-1, in Table 52.

The data in Table 52 show that, with the release conditions postulated in preceeding sections, the BWR site boundary thyroid dose equivalent can be reduced to 10-14 mrem/yr for an average site with two 1,000 MWe units (the lower number being attributed to a 100% organic radioiodine release). However, the assumption of the continual presence of an individual at the site boundary is not realistic, since the nearest residence and dairy farm are not usually located at the same position nor at the site boundary. To illustrate the reduction in dose which may be obtained, Table 52 lists the dose equivalents for the maximum individual at all three locations. It appears that an order of magnitude reduction, or more, is possible as one proceeds from the site boundary to the nearest farm.

Depending on the assumed chemical form of the radioiodine discharged, the critical individual will change from the 6-month-old infant, for the case where the iodine is 100% elemental, to the 4-year-old child for the case where the iodine is 100% organic. This is because for the latter assumption the inhalation pathway prevails, and the combination of internal dose factor and breathing rate for the child results in a high dose equivalent rate for a given air concentration of radioiodine. However, in the realistic case where some iodine is expected to be elemental, and if the assumption is made that a milk pathway exists, the 6-month-old infant will be the critical individual, and the individual thyroid dose equivalent calculated for that age will be limiting.

Worldwide Dose Contributions

In addition to producing local radiation exposure, two long-lived radionuclides are produced which can accumulate in the biosphere and which are capable of migrating over large distances. The two radionuclides of concern are the 12.3-year radioisotope of hydrogen, tritium, and the 10.8-year noble gas, krypton-85. Tritium is released from reactors via liquid effluents as tritiated water, HTO, and in this form enters into the water cycle. The evaporation and subsequent movement of tritiated water vapor permits worldwide dispersion. Krypton-85, however, is a non-reactive gas and diffuses in the atmosphere.

Although the dose contributions from these two radionuclides may be significant in other portions of the fuel cycle, their cumulative effect is much smaller from reactors (Tables 53 and 54) because of the smaller release rates from power stations compared to fuel reprocessing plants. The annual discharge of these two materials from a reactor is approximately equivalent to the daily discharge rate from a spent-fuel reprocessing plant and the resultant exposures are correspondingly smaller for the reactor.

Table 53

Worldwide Health Risk Contributions from Reactor Tritium Releases

TRITIUM TREATMENT OPTION	ANNUAL[a] DISCHARGE (Ci/yr)	TOTAL[a] ANNUAL DOSE COMMITMENT (Man-Rem)	ANNUAL HEALTH[a,d] RISK COMMITMENT		TOTAL HEALTH RISK COMMITMENT 30 YR[b]	TOTAL YEAR 2000 30-YR COMMITMENT[c] FOR ALL U.S. PLANTS	
			WHOLE BODY	THYROID		PERSON-REM	HEALTH EFFECTS
BWR-1,2,3	200	0.18	0.126(-03)	.331(-05)	0.0019	1,080	0.78
BWR-4	130	0.117	0.819(-04)	.215(-05)	0.0013	702	0.50
PWR-1,2	1200	1.08	0.756(-03)	.199(-04)	0.0116	12,960	9.3
PWR-3,4	760	0.684	0.479(-03)	.126(-04)	0.0074	8,208	5.9

Note: $.479(-03) = .479 \times 10^{-3}$ or .000479

[a] for 2 1 GWe plant site

[b] per GWe reactor

[c] 400 GWe BWR; 800 GWe PWR

[d] assuming worldwide age breakdown equivalent to that given for U.S.

Table 54

Worldwide Health Risk From Reactor Krypton-85 Releases

System	Annual Discharge (Ci/yr)[a]	Annual 100-yr Integrated Population Dose Commitment[a] (person-rem)	Annual Health Effects Committed[a]	Commitment from 30-yr 1 GWe Plant Operation		Total Commitment from 30-yr Operation of all U.S. Plants in Year 2000[b]	
				person-rem	health effects	person-rem	health effects
BWR 30-min Holdup & All Delay Systems	760.0	0.266	0.186(-03)	4.0	0.28(-02)	1596.0	1.1
BWR Cryogenic Distillation (Kr DF = 250)	7.8	.0028	.19 (-05)	0.041	.28(-04)	16.5	0.011
BWR Cryogenic Distillation (Kr DF = 2,500)	3.1	.0011	.76 (-06)	.016	.114 (-04)	6.5	.0046
PWR Source Term & All Delay Systems	1600.0	0.56	.392(-03)	8.4	.588(-02)	6720.0	4.70
PWR Cryogenic Distillation (Kr DF = 250)	50.0	.0175	.123(-04)	.262	.184(-03)	210.0	.147
PWR Cryogenic Distillation (Kr DF = 2,500)	45.0	.0158	.110(-04)	.236	.166(-03)	189.0	.142
PWR Cover Gas Recycle	4.4	0.0015	.108(-05)	0.023	.162(-04)	18.5	0.013

[a] Per 2 unit plant.
[b] 400 GWe BWR, 800 GWe PWR.

ECONOMIC AND ENVIRONMENTAL COSTS

Total Costs

The total cost of a treatment system was calculated from the annualized cost as being equivalent to the present worth of the annualized costs using a 7.5% present worth rate. The present worth of all costs is defined as

$$P.W. = \sum_{n=1}^{m} \frac{Pn}{(1+r)^n}$$

where "Pn" is the cost in the nth year and "r" is the applicable interest rate. For the present analysis, $r = 0.075$, $m = 30$ years, and due to the depreciation method assumed in the analysis, the annualized costs, $Pn = P$, are constant so that this term may be removed from the sum:

$$P.W. = P \sum_{n=1}^{30} \frac{1}{(1.075)^n}$$

The present worth factor, which is the sum given in the above equation, is equal to 11.8104 for the parameters given. Thus the total cost of a reactor treatment system is equivalent to 11.8104 times the annualized cost of that system. The total economic impact of a given discharge control option may be determined by multiplying its total cost per nominal GWe reactor by the appropriate number of reactor plants assumed operable by the year 2000 (800 PWRs and 400 BWRs).

The cost to the consumer, which is an important consideration in ascertaining the economic impact of any effluent control measure, was computed from the annualized cost using a capacity factor of 0.8. The conversion factor to mills per kilowatt-hour from annualized dollars per GWe plant is 1.43×10^{-7}.

The environmental impact was determined by multiplying the annual impact per GWe plant by the operating lifetime of the plant (30 years). The total environmental impact for that type of plant was then computed by multiplying the impact for a GWe plant by the number of such plants predicted to be in operation by the year 2000 (800 for PWRs and 400 for PWRs). Included in this estimate are also the worldwide impacts from radionuclides with long half-lives, e.g., krypton-85 among the noble gases, tritium, cobalt-60, strontium-90, cesium-134, and cesium-137 in the liquids, and iodine-131 among the gaseous radioiodines.

The economic and environmental impacts associated with selected effluent control options for an average site are presented in Tables 55-61. These tables illustrate the range of system effectiveness, economic commitments, and environmental commitments that may be achieved for a nominal one GWe plant and for all such plants estimated to exist by the year 2000. The health risks associated with the operation of a single plant are less than six per year, even assuming the hypothetical "zero treatment" options for all release pathways. The annualized cost for the "maximum treatment" options on all release pathways amounts to about $2,800,000 for a BWR and about $1,500,000 for a PWR. The added costs of these treatment systems to the consumer is also small (up to a few tenths of a mill per kilowatt-hour for the most advanced systems).

On a national scale, however, both health risk commitments and control costs become appreciable. With the hypothetical "zero treatment" cases (minimal control costs), total health risks would reach into the neighborhood of 100,000, while minimal health risks (on the order of 25) may be achieved with the expenditure of billions of dollars.

Cost-Effectiveness and the Consumer Perspective

Cost-effectiveness for effluent control systems may be considered from two different but complementary viewpoints: the cost per risk averted or the cost per benefit received. Tables 55-61 detail the cost per risk eliminated by various types of effluent control options applicable to light-water reactors. Except for the radioiodine control options, the cost per risk eliminated increases from about $10,000 - $100,000 (for the first option addition) to well over one million dollars for "maximum treatment" options. Although radioiodine discharge control options are generally more expensive in terms of cost per risk reduced, they more effectively limit the maximum individual dose.

Despite a national total expenditure in excess of 30 billion dollars, the cost of achieving "maximum treatment" for all reactors, the added cost to the consumer per benefit received would be only about 0.38 mills per kilowatt-hour of electricity consumed. When compared to production costs (Table 62) or typical charges for power (Table 63), this cost appears small, making up about 5% of power production costs and even less for typical power charges. For a typical consumer using 7,700 kilowatt-hours per year (based upon average residential comsumption of 5491 kWh in 1968), the total annual cost of achieving even this maximum level of control would amount to $2.92, or less than 25 cents per month.

Although the consumer cost for installation of the maximum control technology appears small on a mills per kilowatt-hour basis, the total cost involved amounts to over 30 billion dollars. Analysis of the total risks reduced after the addition of two or three discharge control options to the various source terms ("zero treatment") generally shows relatively small reductions. Justification for effluent control options is generally difficult to provide since other technical factors (such as maximum individual dose equivalent), as well as political and social considerations, usually enter into the decision-making process.

TABLE 55

BWR Noble Gas Systems: Cost-Effectiveness

System Designation	Dollar Costs Per GWE			Environmental Costs				Cost Effectiveness[a]	
				Annual Per GWE		30-YR Total:All Reactors			
	Annual Cost ($1000)	Consumer Cost (mill/kWh)	Present Worth ($Millions)	Person-Rem	Health Risk	Person-Rem	Health Risk	ΔDollars / ΔHealth Risk	ΔHealth Risk / ΔDollars
CLASS 0[b]	300	0.043	3.043	2910.	1.2(o)[c]	3.5(7)	1.4(4)	1.0(5)	9.8(-6)
CLASS 2-10	600	0.086	7.086	49.7	3.5(-2)	6.0(5)	4.2(2)	1.0(6)	1.0(-6)
CLASS 2-20	665	0.095	7.854	13.3	9.3(-3)	1.6(5)	1.1(2)	4.6(7)	2.2(-8)
CLASS 2-60	955	0.14	11.279	9.7	6.8(-3)	1.2(5)	8.2(1)	9.7(7)	1.0(-8)
CLASS 4	1300	0.19	15.354	7.7	5.4(-3)	9.2(4)	6.5(1)	8.7(6)	1.2(-7)
CLASS 5E	1400	0.20	16.535	1.2	8.5(-4)	1.4(4)	1.0(1)		

[a] Dollars are present worth dollars and health risks are 30 year total; each is placed on a per GWe basis.

[b] Includes 1300 person-rem/yr per GWe and 0.074 health risks/yr per GWe due to radioiodine released at air ejector (assuming that the radioiodine released is 50% elemental and 50% organic). Remaining systems listed do not release this source of radioiodine to environment.

[c] An expression such as 1.2(0) is equivalent to 1.2×10^0, or 1.2.

TABLE 56

PWR Noble Gas Systems: Cost-Effectiveness

System Designation	Dollar Costs Per GWE: Annual Cost ($1000)	Dollar Costs Per GWE: Consumer Cost (mill/kWh)	Dollar Costs Per GWE: Present Worth ($Millions)	Environmental Costs: Annual Per GWE: Person-Rem	Environmental Costs: Annual Per GWE: Health Risk	Environmental Costs: 30-YR Total:All Reactors: Person-Rem	Environmental Costs: 30-YR Total:All Reactors: Health Risk	Cost Effectiveness[a]: ΔDollars / ΔHealth Risk	Cost Effectiveness[a]: ΔHealth Risk / ΔDollars
CLASS 0	0	0.0000	0.000	44.6	3.1(-2)[b]	1.1(6)	7.5(2)		
								4.8(5)	2.1(-6)
CLASS 1A-15	30	0.0042	0.708	9.2	6.4(-3)	2.2(5)	1.5(2)		
								5.9(6)	1.7(-7)
CLASS 1A-60	75	0.011	1.771	4.9	3.4(-3)	1.2(5)	8.2(1)		
								3.1(7)	3.2(-8)
CLASS 2A	300	0.043	7.086	0.8	5.5(-4)	1.9(4)	1.3(1)		
								∞[c]	0
CLASS 3	330	0.047	7.794	0.8	5.5(-4)	1.9(4)	1.3(1)		
								3.0(8)	3.4(-9)
CLASS 5C	405	0.058	9.566	0.7	4.5(-4)	1.6(4)	1.1(1)		

[a] Dollars are present worth dollars and health risks are 30 year totals; each is placed on a per GWe basis.

[b] An expression such as 3.1(-2) is equivalent to 3.1×10^{-2}, or 0.031.

[c] Because Class 3 achieves the same level of health risk as Class 2A (but costs slightly more than Class 2A).

TABLE 57

BWR Radioiodine Systems: Cost-Effectiveness
(Elemental Form)

System Designation	Dollar Costs Per GWE			Environmental Costs				Cost Effectiveness[a]	
				Annual Per GWE		30-YR Total:All Reactors			
	Annual Cost ($1000)	Consumer Costs (mill/kWh)	Present Worth ($Millions)	Person-Rem	Health Risk	Person-Rem	Health Risk	ΔDollars / ΔHealth Risk	ΔHealth Risk / ΔDollars
BGIE-1	0	0.00	0.000	69.5	3.9(-3)[b]	8.3(5)	4.7(1)		
								1.9(7)	5.2(-8)
BGIE-2	150	0.021	1.772	15.3	8.5(-4)	1.8(5)	1.0(1)		
								1.3(8)	7.7(-9)
BGIE-3	375	0.054	4.429	3.0	1.7(-4)	3.6(4)	2.0(0)		
								1.3(9)	7.9(-10)
BGIE-4	600	0.086	7.086	1.8	1.0(-5)	2.2(4)	1.2(0)		
								2.2(9)	4.4(-10)
BGIE-5	800	0.11	9.448	1.2	6.5(-5)	1.4(4)	7.8(-1)		
								2.9(8)	3.4(-9)
BGIE-6	820	0.12	9.685	0.7	3.8(-5)	8.4(3)	4.6(-1)		
								1.7(9)	5.7(-10)
BGIE-7	920	0.13	10.866	0.3	1.6(-5)	3.0(3)	1.9(-1)		

[a] Dollars are present worth dollars and health risks are 30 year total; each is placed on a per GWe basis.

[b] An expression such as 3.9(-3) is equivalent to 3.9×10^{-3}, or 0.0039.

TABLE 58

BWR Radioiodine Systems: Cost-Effectiveness
(Organic Form)

System Designation	Dollar Costs Per GWE			Environmental Costs				Cost Effectiveness[a]	
				Annual Per GWE		30-YR Total:All Reactors			
	Annual Cost ($1000)	Consumer Cost (mill/kWh)	Present Worth ($Millions)	Person-Rem	Health Risk	Person-Rem	Health Risk	ΔDollars / ΔHealth Risk	ΔHealth Risk / ΔDollars
BGIO-1	0	0.000	0.000	119.0	6.5(-3)[b]	1.4(6)	7.8(1)		
BGIO-2	200	0.029	2.362	15.2	8.5(-4)	1.8(6)	1.0(1)	1.4(7)	7.2(-8)
BGIO-3	400	0.057	4.724	6.9	3.9(-4)	8.3(4)	4.6(0)	1.7(8)	5.9(-9)
BGIO-4	420	0.060	4.960	6.3	3.6(-4)	7.6(4)	4.3(0)	2.6(8)	3.8(-9)
BGIO-5	570	0.082	6.732	2.1	1.2(-4)	2.5(4)	1.4(0)	2.5(8)	4.1(-9)
BGIO-6	795	0.11	9.389	1.1	6.5(-5)	1.3(4)	7.8(-1)	1.7(9)	5.9(-10)
BGIO-7	895	0.13	10.570	1.1	6.0(-5)	1.3(4)	7.2(-1)	2.3(9)	4.3(-10)

[a] Dollars are present worth dollars and health risks are 30 year total; each is placed on a per GWe basis.

[b] An expression such as 6.5(-3) is equivalent to 6.5×10^{-3} or 0.0065.

TABLE 59

PWR Radioiodine Systems: Cost-Effectiveness
(Elemental Form)

System Designation	Dollar Costs Per GWE			Environmental Costs				Cost Effectiveness[a]	
				Annual Per GWE		30-YR Total:All Reactors			
	Annual Cost ($1000)	Consumer Costs (mills/ kWh)	Present Worth ($Millions)	Person-Rem	Health Risk	Person-Rem	Health Risk	ΔDollars / ΔHealth Risk	ΔHealth Risk / ΔDollars
PGIE-1	0	0.0000	0.000	97.5	5.5(-3)[b]	2.3(6)	1.3(2)	1.4(7)	7.2(-8)
PGIE-2	60	0.0085	0.709	67.5	3.8(-3)	1.6(6)	9.1(1)	3.8(6)	2.6(-7)
PGIE-3	80	0.011	0.945	31.3	1.8(-3)	7.5(5)	4.2(1)	5.4(7)	1.9(-8)
PGIE-4	230	0.033	2.716	11.4	6.5(-4)	2.7(5)	1.6(1)	5.7(7)	1.8(-8)
PGIE-5	280	0.040	3.307	5.5	3.1(-4)	1.3(5)	7.3(0)	5.4(8)	1.9(-9)
PGIE-6	430	0.061	5.078	3.5	2.0(-4)	8.4(4)	4.7(0)	5.1(8)	1.9(-9)
PGIE-7	580	0.083	6.850	1.5	4.0(-5)	3.5(4)	1.9(0)	7.9(9)	1.3(-10)
PGIE-8	680	0.097	8.031	1.4	3.8(-5)	3.2(4)	1.8(0)		

[a] Dollars are present worth dollars and health risks are 30 year total; each is placed on a per GWe basis.

[b] An expression such as 5.5(-3) is equivalent to $5.5x10^{-3}$, or 0.0055.

TABLE 60

PWR Radioiodine Systems: Cost-Effectiveness (Organic Form)

System Designation	Dollar Costs Per GWE			Environmental Costs				Cost Effectiveness[a]	
				Annual Per GWE		30-YR Total:All Reactors			
	Annual Cost ($1000)	Consumer Cost (mills/kWh)	Present Worth ($Millions)	Person-Rem	Health Risk	Person-Rem	Health Risk	ΔDollars / ΔHealth Risk	ΔHealth Risk / ΔDollars
PGIO-1	0	0.0000	0.000	105.	6.0(-3)[b]	2.5(6)	1.4(2)		
PGIO-2	60	0.0085	0.709	94.5	5.5(-3)	2.3(6)	1.3(2)	4.7(7)	2.1(-8)
PGIO-3	80	0.011	0.945	67.0	3.8(-3)	1.6(6)	9.0(1)	4.5(6)	2.2(-7)
PGIO-4	130	0.019	1.535	44.7	2.5(-3)	1.1(6)	6.0(1)	1.6(7)	6.4(-8)
PGIO-5	280	0.040	3.307	6.8	3.8(-4)	1.6(5)	9.1(0)	2.8(7)	3.6(-8)
PGIO-6	430	0.061	5.078	3.1	1.7(-4)	7.3(4)	4.1(0)	2.8(8)	3.6(-9)
PGIO-7	580	0.083	6.850	3.0	1.7(-4)	7.1(4)	4.0(0)	1.2(10)	8.5(-11)
PGIO-8	680	0.097	8.031	3.0	1.7(-4)	7.1(4)	4.0(0)	∞[c]	0

[a] Dollars are present worth dollars and health risks are 30 year total; each is placed on a per GWe basis.

[b] An expression such as 6.0(-3) is equivalent to 6.0×10^{-3}, or 0.006.

[c] Because PGIO-8 achieves the same level of health risk as PGIO-7 but at greater cost.

TABLE 61

Liquid Systems: Cost-Effectiveness

	Dollar Costs Per GWE			Environmental Costs						Cost-Effectiveness[a]	
				Annual Per GWE			30 YR Total: All Reactors				
System Designation	Annual Cost ($1000)	Consumer Cost (mills/kWh)	Present Worth ($Millions)	Whole Body Person-Rem	Thyroid Person-Rem	Total Health Risk	Whole Body Person-Rem	Thyroid Person-Rem	Total Health Risk	ΔDollars / ΔHealth Risk	ΔHealth Risk / ΔDollars
BWR Systems											
BWR-1	26	0.0037	0.307	1,650	8,000.	1.6(0)[b]	2.0(7)	9.6(7)	1.9(4)		
										8.6(4)	1.2(-5)
BWR-2	61	0.0087	0.720	4.5	80.	7.6(-3)	5.4(4)	9.6(5)	9.2(1)		
										7.8(6)	1.3(-7)
BWR-3	140	0.020	1.65	2.1	37.	3.5(-3)	2.5(4)	4.4(5)	4.4(1)		
										3.4(7)	3.0(-8)
BWR-4	440	0.063	5.19	1.9(-2)	2.9	1.7(-4)	2.3(2)	3.5(4)	2.0(0)		
PWR Systems											
PWR-1	90	0.013	1.06	6,500.	12,500.	5.3(0)	1.6(8)	3.0(8)	1.3(5)		
										8.3(3)	1.2(-4)
PWR-2	200	0.029	2.36	90.	550.	9.4(-2)	2.2(6)	1.3(7)	2.3(3)		
										4.1(5)	2.4(-6)
PWR-3	280	0.040	3.30	24.	6.0	1.7(-2)	5.8(5)	1.4(5)	4.1(2)		
										2.6(6)	3.9(-7)
PWR-4	390	0.056	4.60	6.5(-3)	0.9	5.5(-5)	1.6(2)	2.2(4)	1.3(0)		

[a] Dollars are present worth dollars and health risks are 30 year total; each is placed on a per GWe basis.

[b] An expression such as 1.6(0) is equivalent to 1.6×10^0, or 1.6.

Table 62

Cost of Producing Electric Power

Costs Allocated to	Cost (mills/kWh) Range (Consumption)	 U.S. Average
1968 Actual		
Power Production	6.81-9.28	7.75
Fuel (Included in above)	2.47-3.04	2.47
Transmission	1.56-2.26	1.98
Distribution	4.46-7.71	5.69
Total	12.71-19.25	15.42
1990 Projected (1968 dollars)		
Power Production	10.02-12.09	10.83
Fuel (Included in above)	2.67-3.27	2.86
Transmission	2.11-3.96	2.99
Distribution	2.71-7.03	4.43
Total	15.00-23.08	18.25

Source: Federal Power Commission, "The 1970 National Power Survey, Part I" Table 19.11 page I-19-10 (December 1971).

Table 63

Typical Charges for Electric Power Consumption

Type of Use	Cost (mills/kWh) (1969) Range (based on consumption rate)	Average
Industrial	17.2-21.2	19.2
Commercial	23.7-35.8	29.8
Residential	18.0-40.5	29.7

Source: Federal Power Commission, "Typical Electric Bills - 1969" F.P.C. Washington, D.C. (November 1969).

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